September 2011

SUPPLEMENT 34 TO NUREG-0933,
"RESOLUTION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

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References:	pp. R-1 to R-137, Rev. 23	pp. R-1 to R-138, Rev. 24
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TABLE II

LIST OF ALL THREE MILE ISLAND NUCLEAR PLANT ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, HUMAN FACTORS ISSUES, AND CHERNOBYL ISSUES

This table contains the priority designations for all issues listed in this report. The “Status/Safety Priority Ranking” column notes those issues covered in other issues described in this document. For example, a notation of “I.A.2.2” in the Status/Safety Priority Ranking column for item I.A.2.6(3) means that item I.A.2.6(3) is covered in item I.A.2.2. For resolved issues that resulted in new requirements for operating plants, the appropriate multiplant licensing action number is given in the “MPA No.” column. (The multiplant licensing action numbering system is not related to the numbering systems used to identify the prioritized issues.) This table is maintained primarily for historical purposes.

Legend

ACTIVE	Generic issue that involves actions under the GI Program
DROP	Issue dropped from further pursuit as a generic issue
EI	Environmental issue
I	Resolved TMI Action Plan item with implementation of resolution mandated by NUREG-0737
LI	Licensing issue
MEDIUM	Medium safety priority
MPA	Multiplant action
NA	Not applicable
NOTE 3(a)	Resolution resulted in regulatory products (rule, Standard Review Plan change, or equivalent)
NOTE 3(b)	Resolution did not result in a regulatory product
NOTE 5	Issue that is not a generic safety issue but should be assigned resources for completion. As clarified by SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011, Generic Issues Program will not pursue any further actions toward resolution of Licensing and Regulatory impact issues.
RI	Regulatory impact issue
ROI	Regulatory office implementation: A formal GI for which RES actions of safety/risk assessment or regulatory assessment are complete and remaining actions reside with program offices (e.g., regulatory compliance, reactor oversight process, rulemaking, further research, coordination with industry initiatives)
USI	Unresolved safety issue

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	THREE MILE ISLAND NUCLEAR PLANT ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	-	NRR/DHFS/LQB	I	3	12/31/97	F-01
I.A.1.2	Shift Supervisor Administrative Duties	-	NRR/DHFS/LQB	I	3	12/31/97	
I.A.1.3	Shift Manning	-	NRR/DHFS/LQB	I	3	12/31/97	F-02
I.A.1.4	Long-Term Upgrading	R. Colmar	RES/DFO/HFBR	NOTE 3(a)	3	12/31/97	
I.A.2	Training and Qualifications of Operating Personnel						
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications						
I.A.2.1(1)	Qualifications—Experience	-	NRR/DHFS/LQB	I	6	12/31/97	F-03
I.A.2.1(2)	Training	-	NRR/DHFS/LQB	I	6	12/31/97	F-03
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	-	NRR/DHFS/LQB	I	6	12/31/97	F-03
I.A.2.2	Training and Qualifications of Operations Personnel	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I	6	12/31/97	
I.A.2.4	NRR Participation in Inspector Training	R. Colmar	NRR/DHFS/LQB	LI (NOTE 3)	6	12/31/97	NA
I.A.2.5	Plant Drills	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-			
I.A.2.6(1)	Revise Regulatory Guide 1.8	R. Colmar	NRR/DHFT/HFIB	NOTE 3(a)	6	12/31/97	NA
I.A.2.6(2)	Staff Review of NRR 80-117	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(3)	Revise 10 CFR 55	R. Colmar	NRR/DHFS/LQB	I.A.2.2	6	12/31/97	NA
I.A.2.6(4)	Operator Workshops	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(5)	Develop Inspection Procedures for Training Program	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(6)	Nuclear Power Fundamentals	R. Colmar	NRR/DHFS/LQB	DROP	6	12/31/97	NA
I.A.2.7	Accreditation of Training Institutions	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	R. Emrit	NRR/DHFS/LQB	I	6	12/31/97	
I.A.3.2	Operator Licensing Program Changes	R. Emrit	NRR/DHFS/OLB	NOTE 3(b)	6	12/31/97	NA
I.A.3.3	Requirements for Operator Fitness	R. Colmar	RES/DRAO/HFSB	NOTE 3(b)	6	12/31/97	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.A.3.4	Licensing of Additional Operations Personnel	D. Thatcher	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	D. Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	6	12/31/97	NA
<u>I.A.4</u>	<u>Simulator Use and Development</u>						
I.A.4.1	Initial Simulator Improvement						
I.A.4.1(1)	Short-Term Study of Training Simulators	D. Thatcher	NRR/DHFS/OLB	NOTE 3(b)	6	12/31/97	NA
I.A.4.1(2)	Interim Changes in Training Simulators	D. Thatcher	NRR/DHFS/OLB	NOTE 3(a)	6	12/31/97	
I.A.4.2	Long-Term Training Simulator Upgrade						
I.A.4.2(1)	Research on Training Simulators	R. Colmar	NRR/DHFT/HFIB	NOTE 3(a)	6	12/31/97	
I.A.4.2(2)	Upgrade Training Simulator Standards	R. Colmar	RES/DFO/HFBR	NOTE 3(a)	6	12/31/97	
I.A.4.2(3)	Regulatory Guide on Training Simulators	R. Colmar	RES/DFO/HFBR	NOTE 3(a)	6	12/31/97	
I.A.4.2(4)	Review Simulators for Conformance to Criteria	R. Colmar	NRR/DLPQ/LOLB	NOTE 3(a)	6	12/31/97	
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	R. Colmar	RES/DAE/RSRB	LI (NOTE 3)	6	12/31/97	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	R. Colmar	RES/DAE/RSRB	LI (NOTE 3)	6	12/31/97	NA
<u>I.B.</u>	<u>SUPPORT PERSONNEL</u>						
<u>I.B.1</u>	<u>Management for Operations</u>						
I.B.1.1	Organization and Management Long-Term Improvements						
I.B.1.1(1)	Prepare Draft Criteria	R. Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(2)	Prepare Commission Paper	R. Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	R. Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	R. Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	R. Colmar	OIE/DQASIP/ORPB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	R. Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	4	12/31/97	NA
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	R. Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	4	12/31/97	NA
I.B.1.2	Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants						
I.B.1.2(1)	Prepare Draft Criteria	-	NRR/DHFS/LQB	NOTE 3(b)	4	12/31/97	NA
I.B.1.2(2)	Review Near-Term Operating License Facilities	-	NRR/DHFS/LQB	NOTE 3(b)	4	12/31/97	NA
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	-	NRR/DL/ORAB	NOTE 3(b)	4	12/13/97	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.B.1.3	Loss of Safety Function						
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	G. Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	G. Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.1.3(3)	Use Nonfiscal Approaches to Accomplish Safest Shutdown Cooling	G. Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.2	Inspection of Operating Reactors						
I.B.2.1	Revise OIE Inspection Program						
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(3)	Followup on Completed Maintenance Work Orders to Assure Proper Testing and Return to Service	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(4)	Observe Surveillance Tests to Determine whether Test Instruments Are Properly Calibrated	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(6)	Observe Routine Maintenance	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.2	Resident Inspector at Operating Reactors	G. Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.3	Regional Evaluations	G. Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.4	Overview of Licensee Performance	G. Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision						
I.C.1(1)	Small Break LOCAs	-	NRR	I	4	12/31/97	
I.C.1(2)	Inadequate Core Cooling	-	NRR	I	4	12/31/97	F-04
I.C.1(3)	Transients and Accidents	-	NRR	I	4	12/31/97	F-05
I.C.1(4)	Confirmatory Analyses of Selected Transients	R. Riggs	NRR/DSI/RSB	NOTE 3(b)	4	12/31/97	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.C.2	Shift and Relief Turnover Procedures	-	NRR	I	4	12/31/97	
I.C.3	Shift Supervisor Responsibilities	-	NRR	I	4	12/31/97	
I.C.4	Control Room Access	-	NRR	I	4	12/31/97	
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	-	NRR/DL	I	4	12/31/97	F-06
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	-	NRR/DL	I	4	12/31/97	F-07
I.C.7	NSSS Vendor Review of Procedures	-	NRR/DHFS/PSRB	I	4	12/31/97	
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR/DHFS/PSRB	I	4	12/31/97	
I.C.9	Long-Term Program Plan for Upgrading of Procedures	R. Riggs	NRR/DHFS/PSRB	NOTE 3(b)	4	12/31/97	NA
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	-	NRR/DL	I	8	12/31/97	F-08
I.D.2	Plant Safety Parameter Display Console	-	NRR/DL	I	8	12/31/97	F-09
I.D.3	Safety System Status Monitoring	D. Thatcher	RES/DE/MEB	NOTE 3(b)	8	12/31/97	NA
I.D.4	Control Room Design Standard	D. Thatcher	RES/DRPS/RHFB	NOTE 3(b)	8	12/31/97	NA
I.D.5	Improved Control Room Instrumentation Research						
I.D.5(1)	Operator-Process Communication	D. Thatcher	RES/DFO/HFBR	NOTE 3(b)	8	12/31/97	NA
I.D.5(2)	Plant Status and Post-Accident Monitoring	D. Thatcher	RES/DFO/HFBR	NOTE 3(a)	8	12/31/97	
I.D.5(3)	On-Line Reactor Surveillance System	D. Thatcher	RES/DE/MEB	NOTE 3(b)	8	12/31/97	NA
I.D.5(4)	Process Monitoring Instrumentation	D. Thatcher	RES/DFO/ICBR	NOTE 3(b)	8	12/31/97	NA
I.D.5(5)	Disturbance Analysis Systems	D. Thatcher	RES/DRPS/RHFB	LI (NOTE 3)	8	12/31/97	NA
I.D.6	Technology Transfer Conference	D. Thatcher	RES/DFO/HFBR	LI (NOTE 3)	8	12/31/97	NA
I.E	ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE						
I.E.1	Office for Analysis and Evaluation of Operational Data	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.2	Program Office Operational Data Evaluation	P. Matthews	NRR/DL/ORAB	LI (NOTE 3)	3	12/31/97	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.E.3	Operational Safety Data Analysis	P. Matthews	RES/DRA/RRBR	LI (NOTE 3)	3	12/31/97	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.5	Nuclear Plant Reliability Data System	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.6	Reporting Requirements	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.7	Foreign Sources	P. Matthews	IP	LI (NOTE 3)	3	12/31/97	NA
I.E.8	Human Error Rate Analysis	P. Matthews	RES/DFO/HFBR	LI (NOTE 3)	3	12/31/97	NA
I.F	QUALITY ASSURANCE						
I.F.1	Expand QA List	J. Pittman	RES/DRA/ARGIB	NOTE 3(b)	4	12/31/98	NA
I.F.2	Develop More Detailed QA Criteria						
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	J. Pittman	OIE/DQASIP/QUAB	DROP	5	09/30/11	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	5	09/30/11	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	5	09/30/11	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	J. Pittman	OIE/DQASIP/QUAB	DROP	5	09/30/11	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	J. Pittman	OIE/DQASIP/QUAB	DROP	5	09/30/11	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	5	09/30/11	NA
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	J. Pittman	OIE/DQASIP/QUAB	DROP	5	09/30/11	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	J. Pittman	OIE/DQASIP/QUAB	DROP	5	09/30/11	NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	5	09/30/11	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	J. Pittman	OIE/DQASIP/QUAB	DROP	5	09/30/11	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	J. Pittman	OIE/DQASIP/QUAB	DROP	5	09/30/11	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I	3	12/31/97	
I.G.2	Scope of Test Program	H. Vandermolen	NRR/DHFS/PSRB	NOTE 3(a)	3	12/31/97	NA
II.A	SITING						
II.A.1	Siting Policy Reformulation	H. Vandermolen	NRR/DE/SAB	NOTE 3(b)	2	12/31/97	NA
II.A.2	Site Evaluation of Existing Facilities	H. Vandermolen	NRR/DE/SAB	V.A.1	2	12/31/97	NA
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I	5	09/30/11	F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	- -	NRR/DL	I	5	09/30/11	F-11
II.B.3	Post-Accident Sampling		NRR/DL	I	5	09/30/11	F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I	5	09/30/11	F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting						
II.B.5(1)	Behavior of Severely Damaged Fuel	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	5	09/30/11	NA
II.B.5(2)	Behavior of Core-Melt	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	5	09/30/11	NA
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	5	09/30/11	NA
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	J. Pittman	NRR/DST/RRAB	NOTE 3(a)	5	09/30/11	
II.B.7	Analysis of Hydrogen Control	P. Matthews	NRR/DSI/CSB	II.B.8	5	09/30/11	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	H. Vandermolen	RES/DRAO/RAMR	NOTE 3(a)	5	09/30/11	
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	J. Pittman	RES/DRAO/RRB	NOTE 3(b)	3	12/31/97	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	J. Pittman	NRR/DST/RRAB	NOTE 3(b)	3	12/31/97	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.C.3	Systems Interaction	J. Pittman	NRR/DST/GIB	A-17	3	12/31/97	NA
II.C.4	Reliability Engineering	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	3	12/31/97	NA
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I	3	12/31/98	F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	R. Riggs	RES	DROP	3	12/31/98	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I	3	12/31/98	
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I	2	12/31/97	F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I	2	12/31/97	F-16, F-17
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	R. Riggs	RES/DRA/RRBR	NOTE 3(a)	2	12/31/97	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	R. Riggs	NRR/DSI/RSB	II.K.3(17)	3	12/31/98	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	R. Riggs	RES/DAE/RSRB	NOTE 3(b)	3	12/31/98	NA
II.E.2.3	Uncertainties in Performance Predictions	H. Vandermolen	NRR/DSI/RSB	DROP	3	12/31/98	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR/DL	I	2	12/31/97	
II.E.3.2	Systems Reliability	H. Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	H. Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.4	Alternate Concepts Research	R. Riggs	RES/DAE/FBRB	NOTE 3(b)	2	12/31/97	NA
II.E.3.5	Regulatory Guide	R. Riggs	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I	2	12/31/97	F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I	2	12/31/97	F-19
II.E.4.3	Integrity Check	W. Milstead	RES/DRPS/RPSI	NOTE 3(b)	2	12/31/97	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.E.4.4	Purging						
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(4)	Evaluate Purging and Venting during Normal Operation	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	D. Thatcher	NRR/DSI/RSB	NOTE 3(a)	2	12/31/98	
II.E.5.2	B&W Reactor Transient Response Task Force	D. Thatcher	NRR/DL/ORAB	NOTE 3(a)	2	12/31/98	
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	D. Thatcher	RES/DE/EIB	NOTE 3(a)	2	12/31/98	
II.F	INSTRUMENTATION AND CONTROLS						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I	3	12/31/98	F-20, F-21, F-22, F-23, F-24, F-25
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I	3	12/31/98	F-26
II.F.3	Instruments for Monitoring Accident Conditions	H. Vandermolen	RES/DFO/ICBR	NOTE 3(a)	3	12/31/98	
II.F.4	Study of Control and Protective Action Design Requirements	D. Thatcher	NRR/DSI/ICSB	DROP	3	12/31/98	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	D. Thatcher	RES/DE	LI (NOTE 3)	3	12/31/98	NA
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I	1	12/31/98	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	P. Matthews	NRR/TMIPO	NOTE 3(b)	3	12/31/98	NA
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	W. Milstead	RES/DRAA/AEB	NOTE 3(b)	3	12/31/98	NA
II.H.3	Evaluate and Feed Back Information Obtained from TMI	W. Milstead	NRR/TMIPO	II.H.2	3	12/31/98	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	W. Milstead	RES/DHSWM/SEBR	LI (NOTE 3)	3	12/31/98	NA
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.2	Modify Existing Vendor Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.3	Management for Design and Construction						
II.J.3.1	Organization and Staffing to Oversee Design and Construction	J. Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.3.2	Issue Regulatory Guide	J. Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	L. Riani	AEOD/DSP/ROAB	NOTE 3(a)	3	12/31/98	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins						
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(4)	Review Operating Procedures and Training Instructions	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(5)	Safety-Related Valve Position Description	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred Out of Containment Inadvertently	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(12)	One-Hour Notification Requirement and Continuous Communications Channels	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(17)	Trip PZR Level Bistable So That PZR Low Pressure Will Initiate Safety Injection	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses between Reactor Trip and RCP Trip	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(25)	Develop Operator Action Guidelines	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.2	Commission Orders on B&W Plants						
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	R. Emrit	NRR/DHFS/OLB	NOTE 3(a)	-	12/31/84	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(7)	Reevaluate Transient of September 24, 1977	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(8)	Continued Upgrading of AFW System	R. Emrit	NRR	II.E.1.1, II.E.1.2	-	12/31/84	NA
II.K.2(9)	Analysis and Upgrading of Integrated Control System	R. Emrit	NRR	I	-	12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	R. Emrit	NRR	I	-	12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	R. Emrit	NRR	I	-	12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA with No AFW	R. Emrit	NRR	I	-	12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	R. Emrit	NRR	I	-	12/31/84	F-31
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes after Primary System Voiding	R. Emrit	NRR	I	-	12/31/84	
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power	R. Emrit	NRR	I	-	12/31/84	F-32
II.K.2(17)	Analysis of Potential Voiding in RCS during Anticipated Transients	R. Emrit	NRR	I	-	12/31/84	F-33
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	R. Emrit	NRR	I	-	12/31/84	F-34
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	R. Emrit	NRR	I	-	12/31/84	F-35
II.K.2(21)	LOFT L3-1 Predictions	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3	Final Recommendations of Bulletins and Orders Task Force						
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	R. Emrit	NRR	I	-	12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	R. Emrit	NRR	I	-	12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	R. Emrit	NRR	I	-	12/31/84	F-38
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	R. Emrit	NRR	II.C.1, II.C.2, II.C.3	-	12/31/84	NA
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	R. Emrit	NRR	I	-	12/31/84	F-39, G-01
II.K.3(6)	Instrumentation to Verify Natural Circulation	R. Emrit	NRR/DSI	I.C.1(3), II.F.2, II.F.3	-	12/31/84	NA
II.K.3(7)	Evaluation of PORV Opening Probability during Overpressure Transient	R. Emrit	NRR	I	-	12/31/84	
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	R. Emrit	NRR/DST/GIB	II.C.1, II.E.3.3	-	12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller Modification	R. Emrit	NRR	I	-	12/31/84	F-40
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	R. Emrit	NRR	I	-	12/31/84	F-41
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc., Until Further Review Complete	R. Emrit	NRR	I	-	12/31/84	
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	R. Emrit	NRR	I	-	12/31/84	F-42
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	R. Emrit	NRR	I	-	12/31/84	F-43
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	R. Emrit	NRR	I	-	12/31/84	F-44
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	R. Emrit	NRR	I	-	12/31/84	F-45

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves—Feasibility Study and System Modification	R. Emrit	NRR	I	-	12/31/84	F-46
II.K.3(17)	Report on Outage of ECC Systems—Licensee Report and Technical Specification Changes	R. Emrit	NRR	I	-	12/31/84	F-47
II.K.3(18)	Modification of ADS Logic—Feasibility Study and Modification for Increased Diversity for Some Event Sequences	R. Emrit	NRR	I	-	12/31/84	F-48
II.K.3(19)	Interlock on Recirculation Pump Loops	R. Emrit	NRR	I	-	12/31/84	F-49
II.K.3(20)	Loss of Service Water for Big Rock Point	R. Emrit	NRR	I	-	12/31/84	
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level—Design and Modification	R. Emrit	NRR	I	-	12/31/84	F-50
II.K.3(22)	Automatic Switchover of RCIC System Suction—Verify Procedures and Modify Design	R. Emrit	NRR	I	-	12/31/84	F-51
II.K.3(23)	Central Water Level Recording	R. Emrit	NRR	I.D.2, III.A.1.2(1), III.A.3.4	-	12/31/84	NA
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	R. Emrit	NRR	I	-	12/31/84	F-52
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	R. Emrit	NRR	I	-	12/31/84	F-53
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	R. Emrit	NRR/DSI	II.E.2.1	-	12/31/84	NA
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	R. Emrit	NRR	I	-	12/31/84	F-54
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	R. Emrit	NRR	I	-	12/31/84	F-55
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	R. Emrit	NRR	I	-	12/31/84	F-56
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	R. Emrit	NRR	I	-	12/31/84	F-57
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	R. Emrit	NRR	I	-	12/31/84	F-58
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	R. Emrit	NRR/DSI	II.E.2.2	-	12/31/84	NA
II.K.3(33)	Evaluate Elimination of PORV Function	R. Emrit	NRR	II.C.1	-	12/31/84	NA
II.K.3(34)	Relap-4 Model Development	R. Emrit	NRR/DSI	II.E.2.2	-	12/31/84	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage during a Small-Break LOCA	R. Emrit	NRR	II.K.2(16)	-	12/31/84	NA
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensible Gases	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	R. Emrit	NRR	II.K.2(15)	-	12/31/84	NA
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	R. Emrit	NRR	I	-	12/31/84	F-59
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	R. Emrit	NRR	I	-	12/31/84	F-60
II.K.3(46)	Response to List of Concerns from ACRS Consultant	R. Emrit	NRR	I	-	12/31/84	F-61
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	R. Emrit	NRR	I.C.1(3), II.E.2.2	-	12/31/84	NA
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	R. Emrit	NRR	II.C.1, II.C.2	-	12/31/84	NA
II.K.3(49)	Review of Procedures (NRC)	R. Emrit	NRR/DHFS/PSRB	I.C.8, I.C.9	-	12/31/84	NA
II.K.3(50)	Review of Procedures (NSSS Vendors)	R. Emrit	NRR/DHFS/PSRB	I.C.7,	-	12/31/84 I.C.9	NA
II.K.3(51)	Symptom-Based Emergency Procedures	R. Emrit	NRR/DHFS/PSRB	I.C.9	-	12/31/84	NA
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	R. Emrit	NRR	I.B.1.1, I.C.2, I.C.5	-	12/31/84	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3(53)	Two Operators in Control Room	R. Emrit	NRR	I.A.1.3	-	12/31/84	NA
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	R. Emrit	NRR	I.A.4.1(2)	-	12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	R. Emrit	NRR	I.C.1(3), I.D.2, I.D.3	-	12/31/84	NA
II.K.3(56)	Simulator Training Requirements	R. Emrit	NRR/DHFS/OLB	I.A.2.6(3), I.A.3.1	-	12/31/84	NA
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	R. Emrit	NRR	I	-	12/31/84	F-62
<u>III.A</u>	<u>EMERGENCY PREPAREDNESS AND RADIATION EFFECTS</u>						
<u>III.A.1</u>	<u>Improve Licensee Emergency Preparedness—Short-Term</u>						
III.A.1.1	Upgrade Emergency Preparedness						
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB	I	2	06/30/91	
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.2	Upgrade Licensee Emergency Support Facilities						
III.A.1.2(1)	Technical Support Center	-	OIE/DEPER/EPB	I	2	06/30/91	F-63
III.A.1.2(2)	On-Site Operational Support Center	-	OIE/DEPER/EPB	I	2	06/30/91	F-64
III.A.1.2(3)	Near-Site Emergency Operations Facility	-	OIE/DEPER/EPB	I	2	06/30/91	F-65
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent						
III.A.1.3(1)	Workers	R. Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.3(2)	Public	R. Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
<u>III.A.2</u>	<u>Improving Licensee Emergency Preparedness—Long-Term</u>						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E						
III.A.2.1(1)	Publish Proposed Amendments to the Rules	-	RES	NOTE 3(a)	-	12/31/94	NA
III.A.2.1(2)	Conduct Public Regional Meetings	-	RES	NOTE 3(b)	-	12/31/94	NA
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	-	RES	NOTE 3(b)	-	12/31/94	NA
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	-	OIE	I	-		F-67
III.A.2.2	Development of Guidance and Criteria	-	NRR/DL	I	-		F-68

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.1	NRC Role in Responding to Nuclear Emergencies						
III.A.3.1(1)	Define NRC Role in Emergency Situations	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(4)	Prepare Commission Paper	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.2	Improve Operations Centers	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.3	Communications						
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	J. Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	J. Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.4	Nuclear Data Link	D. Thatcher	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	
III.A.3.5	Training, Drills, and Tests	J. Pittman	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6	Interaction of NRC and Other Agencies						
III.A.3.6(1)	International	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(2)	Federal	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(3)	State and Local	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.B	EMERGENCY PREPAREDNESS OF STATE AND LOCAL GOVERNMENTS						
III.B.1	Transfer of Responsibilities to FEMA	W. Milstead	OIE/DEPER/IRDB	NOTE 3(b)	-	11/30/83	NA
III.B.2	Implementation of NRC and FEMA Responsibilities						
III.B.2(1)	The Licensing Process	W. Milstead	OIE/DEPER/IRDB	NOTE 3(b)	-	11/30/83	NA
III.B.2(2)	Federal Guidance	W. Milstead	OIE/DEPER/IRDB	NOTE 3(b)	-	11/30/83	NA
III.C	PUBLIC INFORMATION						
III.C.1	Have Information Available for the News Media and the Public						
III.C.1(1)	Review Publicly Available Documents	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.C.1(2)	Recommend Publication of Additional Information	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.C.1(3)	Program of Seminars for News Media Personnel	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.C.2	Develop Policy and Provide Training for Interfacing with the News Media						
III.C.2(1)	Develop Policy and Procedures for Dealing with Briefing Requests	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.C.2(2)	Provide Training for Members of the Technical Staff	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure						
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	-	NRR	I	1	12/31/88	
III.D.1.1(2)	Review Information on Provisions for Leak Detection	R. Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	R. Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.2	Radioactive Gas Management	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3	Ventilation System and Radioiodine Adsorber Criteria	-	-	-			
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(2)	Review and Revise SRP	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	R. Emrit	NRR/DSI/METB	NOTE 3(b)	1	12/31/88	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.2	Public Radiation Protection Improvement						
III.D.2.1	Radiological Monitoring of Effluents						
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	R. Emrit	NRR/DSI/METB	DROP	4	09/30/11	NA
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble	R. Emrit	NRR/DSI/METB	DROP	4	09/30/11	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Gases and Radioiodine Released to the Atmosphere						
III.D.2.1(3)	Revise Regulatory Guides	R. Emrit	NRR/DSI/METB	DROP	4	09/30/11	NA
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis						
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	R. Emrit	NRR/DSI/RAB	NOTE 3(b)	4	09/30/11	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	R. Emrit	NRR/DSI/RAB	III.D.2.5	4	09/30/11	NA
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	R. Emrit	NRR/DSI/RAB	III.D.2.5	4	09/30/11	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	R. Emrit	NRR/DSI/RAB	III.D.2.5	4	09/30/11	NA
III.D.2.3	Liquid Pathway Radiological Control						
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	4	09/30/11	NA
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	4	09/30/11	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	4	09/30/11	NA
III.D.2.3(4)	Prepare a Summary Assessment	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	4	09/30/11	NA
III.D.2.4	Offsite Dose Measurements						
III.D.2.4(1)	Study Feasibility of Environmental Monitors	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	4	09/30/11	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	H. Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	4	09/30/11	NA
III.D.2.5	Offsite Dose Calculation Manual	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	4	09/30/11	NA
III.D.2.6	Independent Radiological Measurements	H. Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	4	09/30/11	NA
<u>III.D.3</u>	<u>Worker Radiation Protection Improvement</u>						
III.D.3.1	Radiation Protection Plans	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/87	NA
III.D.3.2	Health Physics Improvements						
III.D.3.2(1)	Amend 10 CFR 20	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(2)	Issue a Regulatory Guide	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(3)	Develop Standard Performance Criteria	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.3	In-plant Radiation Monitoring						
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	-	NRR/DL	I	2	12/31/86	F-69
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for	-	NRR	NOTE 3(a)	2	12/31/86	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Additional Survey Equipment						
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(4)	Issue a Regulatory Guide	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.4	Control Room Habitability	-	NRR/DL	I	2	12/31/86	F-70
III.D.3.5	Radiation Worker Exposure						
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(3)	Revise 10 CFR 20	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
IV.A	STRENGTHEN ENFORCEMENT PROCESS						
IV.A.1	Seek Legislative Authority	R. Emrit	GC	LI (NOTE 3)	-	11/30/83	NA
IV.A.2	Revise Enforcement Policy	R. Emrit	OIE/ES	LI (NOTE 3)	-	11/30/83	NA
IV.B	ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	R. Emrit	OIE/DEPER	LI (NOTE 3)	-	11/30/83	NA
IV.C	EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS						
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	R. Emrit	NMSS/WM	NOTE 3(b)	-	11/30/83	NA
IV.D	NRC STAFF TRAINING						
IV.D.1	NRC Staff Training	R. Emrit	ADM/MDTS	LI (NOTE 3)	-	11/30/83	NA
IV.E	SAFETY DECISION-MAKING						
IV.E.1	Expand Research on Quantification of Safety	R. Colmar	RES/DRA/RABR	LI (NOTE 3)	3	09/30/11	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Decisionmaking						
IV.E.2	Plan for Early Resolution of Safety Issues	R. Emrit	NRR/DST/SPEB	LI (NOTE 3)	3	09/30/11	NA
IV.E.3	Plan for Resolving Issues at the CP Stage	R. Colmar	RES/DRA/RABR	LI (NOTE 5)	3	09/30/11	NA
IV.E.4	Resolve Generic Issues by Rulemaking	R. Colmar	RES/DRA/RABR	LI (NOTE 3)	3	09/30/11	NA
IV.E.5	Assess Currently Operating Reactors	P. Matthews	NRR/DL/SEPB	NOTE 3(b)	3	09/30/11	NA
IV.F	FINANCIAL DISINCENTIVES TO SAFETY						
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	D. Thatcher	OIE/DQASIP	NOTE 3(b)	1	12/31/86	NA
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	P. Matthews	SP	NOTE 3(b)	1	12/31/86	NA
IV.G	IMPROVE SAFETY RULEMAKING PROCEDURES						
IV.G.1	Develop a Public Agenda for Rulemaking	R. Emrit	ADM/RPB	LI (NOTE 3)	1	12/31/86	NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	W. Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.3	Improve Rulemaking Procedures	W. Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	W. Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.H	NRC PARTICIPATION IN THE RADIATION POLICY COUNCIL						
IV.H.1	NRC Participation in the Radiation Policy Council	G. Sege	RES/DHSWM/HEBR	LI (NOTE 3)	-	11/30/83	NA
V.A	DEVELOPMENT OF SAFETY POLICY						
V.A.1	Develop NRC Policy Statement on Safety	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.B	POSSIBLE ELIMINATION OF NONSAFETY RESPONSIBILITIES						
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
V.C	ADVISORY COMMITTEES						
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.C.2	Study Need for Additional Advisory Committees	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.D	LICENSING PROCESS						
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	R. Emrit	GC	LI (NOTE 3)	1	09/30/11	NA
V.D.2	Study Construction-during-Adjudication Rules	R. Emrit	GC	LI (NOTE 5)	1	09/30/11	NA
V.D.3	Reexamine Commission Role in Adjudication	R. Emrit	GC	LI (NOTE 5)	1	09/30/11	NA
V.D.4	Study the Reform of the Licensing Process	R. Emrit	GC	LI (NOTE 5)	1	09/30/11	NA
V.E	LEGISLATIVE NEEDS						
V.E.1	Study the Need for TMI-Related Legislation	R. Emrit	GC	LI (NOTE 5)	1	09/30/11	NA
V.F	ORGANIZATION AND MANAGEMENT						
V.F.1	Study NRC Top Management Structure and Process	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.F.2	Reexamine Organization and Functions of the NRC Offices	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.F.3	Revise Delegations of Authority to Staff	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.G	CONSOLIDATION OF NRC LOCATIONS						
V.G.1	Achieve Single Location, Long-Term	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.G.2	Achieve Single Location, Interim	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-10
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-4	CE Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-5	B&W Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-6	Mark I Short-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-7	Mark I Long-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-01
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-9	ATWS (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-10	BWR Feedwater Nozzle Cracking (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-25
A-11	Reactor Vessel Materials Toughness (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-13	Snubber Operability Assurance	R. Emrit	NRR/DE/MEB	NOTE 3(a)	1	06/30/91	B-17, B-22
A-14	Flaw Detection	P. Matthews	NRR/DE/MTEB	DROP	-	11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	J. Pittman	NRR/DE/CHEB	NOTE 3(b)	-	11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	R. Emrit	NRR/DSI/CPB	NOTE 3(a)	-	11/30/83	D-12
A-17	Systems Interactions in Nuclear Power Plants (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	1	12/31/89	NA
A-18	Pipe Rupture Design Criteria	R. Emrit	NRR/DE/MEB	DROP	-	11/30/83	NA
A-19	Digital Computer Protection System	W. Milstead	RES/DSR/HFB	LI (NOTE 5)	3	09/30/11	NA
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI (NOTE 5)	2	09/30/11	NA
A-21	Main Steamline Break Inside Containment—Evaluation of Environmental Conditions for Equipment Qualification	H. Vandermolen	NRR/DSI/CSB	DROP	1	12/31/98	NA
A-22	PWR Main Steamline Break—Core, Reactor Vessel, and Containment Building Response	H. Vandermolen	NRR/DSI/CSB	DROP	-	11/30/83	NA
A-23	Containment Leak Testing	P. Matthews	NRR/DSI/CSB	RI (NOTE 5)	1	09/30/11	
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-60

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
A-25	Non-Safety Loads on Class 1E Power Sources	D. Thatcher	NRR/DSI/PSB	NOTE 3(a)	-	11/30/83	
A-26	Reactor Vessel Pressure Transient Protection (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI (NOTE 5)	1	09/30/11	NA
A-28	Increase in Spent Fuel Pool Storage Capacity	R. Colmar	NRR/DE/SGEB	NOTE 3(a)	-	11/30/83	
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	R. Colmar	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/89	NA
A-30	Adequacy of Safety-Related DC Power Supplies	G. Sege	NRR/DSI/PSB	128	1	12/31/86	NA
A-31	RHR Shutdown Requirements (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-32	Missile Effects	J. Pittman	NRR/DE/MTEB	A-37, A-38, B-68	-	11/30/83	NA
A-33	NEPA Review of Accident Risks	-	NRR/DSI/AEB	EI (NOTE 3)	-	11/30/83	NA
A-34	Instruments for Monitoring Radiation and Process Variables during Accidents	H. Vandermolen	NRR/DSI/ICSB	II.F.3	-	11/30/83	NA
A-35	Adequacy of Offsite Power Systems	R. Emrit	NRR/DSI/PSB	NOTE 3(a)	1	12/31/94	B-23
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	R. Emrit	NRR/DSI/GIB	NOTE 3(a)	2	06/30/04	C-10, C-15
A-37	Turbine Missiles	J. Pittman	NRR/DE/MTEB	DROP	-	11/30/83	NA
A-38	Tornado Missiles	G. Sege	NRR/DSI/ASB	DROP	3	06/30/00	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-40	Seismic Design Criteria (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	NA
A-41	Long-Term Seismic Program	L. Riani	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-05
A-43	Containment Emergency Sump Performance (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	12/31/87	
A-44	Station Blackout (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(a)	1	06/30/88	
A-45	Shutdown Decay Heat Removal Requirements (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/88	NA
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	R. Emrit	NRR/DSRO/EIB	NOTE 3(a)	2	06/30/00	
A-47	Safety Implications of Control Systems (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	R. Emrit	NRR/DSIR/SAIB	NOTE 3(a)	1	06/30/89	
A-49	Pressurized Thermal Shock (former USI)	R. Emrit	NRR/DSRO/RSIB	NOTE 3(a)	1	12/31/87	A-21
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	EI (NOTE 3)	-	11/30/83	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-3	Event Categorization	-	NRR/DSI/RSB	LI (NOTE 3)	-	11/30/83	NA
B-4	ECCS Reliability	R. Emrit	NRR/DSI/RSB	II.E.3.2	-	11/30/83	NA
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	D. Thatcher	RES/DE/EIB	NOTE 3(b)	1	06/30/88	NA
B-6	Loads, Load Combinations, Stress Limits	J. Pittman	NRR/DSRO/EIB	119.1	-	12/31/87	NA
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (NOTE 3)	-	11/30/83	NA
B-8	Locking Out of ECCS Power Operated Valves	R. Riggs	NRR/DSI/RSB	DROP	1	12/31/94	NA
B-9	Electrical Cable Penetrations of Containment	R. Emrit	NRR/DSI/PSB	NOTE 3(b)	-	11/30/83	NA
B-10	Behavior of BWR Mark III Containments	H. Vandermolen	NRR/DSI/CSB	NOTE 3(a)	1	12/31/84	NA
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI (NOTE 5)	1	09/30/11	NA
B-12	Containment Cooling Requirements (Non-LOCA)	R. Emrit	NRR/DSI/CSB	NOTE 3(b)	1	12/31/86	NA
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI (NOTE 5)	1	09/30/11	NA
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	R. Emrit	NRR/DST/GIB	A-48	-	11/30/83	NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (NOTE 3)	-	11/30/83	NA
B-16	Protection against Postulated Piping Failures in Fluid Systems Outside Containment	R. Emrit	NRR/DE/MEB	A-18	-	11/30/83	NA
B-17	Criteria for Safety-Related Operator Actions	W. Milstead	RES/DST/CIHFB	NOTE 3(b)	3	06/30/00	
B-18	Vortex Suppression Requirements for Containment Sumps	R. Emrit	NRR/DST/GIB	A-43	-	11/30/83	NA
B-19	Thermal-Hydraulic Stability	L. Riani	NRR/DSI/CPB	NOTE 3(b)	-	06/30/85	NA
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI (NOTE 5)	1	09/30/11	
B-21	Core Physics	-	NRR/DSI/CPB	LI (NOTE 3)	-	11/30/83	NA
B-22	LWR Fuel	R. Emrit	RES/DSIR/RPSIB	DROP	2	06/30/95	NA
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (NOTE 3)	-	11/30/83	NA
B-24	Seismic Qualification of Electrical and Mechanical Equipment	R. Emrit	NRR	A-46	-	11/30/83	NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI (NOTE 5)	1	09/30/11	
B-26	Structural Integrity of Containment Penetrations	R. Riggs	NRR/DE/MTEB	NOTE 3(b)	1	12/31/84	NA
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI (NOTE 5)	1	09/30/11	
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-29	Effectiveness of Ultimate Heat Sinks	J. Pittman	NRR/DE/EHEB	LI (NOTE 3)	1	06/30/91	NA
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI (NOTE 5)	1	09/30/11	
B-31	Dam Failure Model	W. Milstead	NRR/DE/SGEB	LI (NOTE 3)	1	06/30/89	NA
B-32	Ice Effects on Safety-Related Water Supplies	J. Pittman	NRR/DE/EHEB	153	1	06/30/91	NA
B-33	Dose Assessment Methodology	-	NRR/DSI/RAB	LI (NOTE 3)	-	11/30/83	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-34	Occupational Radiation Exposure Reduction	R. Emrit	NRR/DSI/RAB	III.D.3.1	-	11/30/83	NA
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	-	NRR/DSI/METB	LI (NOTE 5)	1	09/30/11	
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	R. Emrit	NRR/DSI/METB	NOTE 3(a)	-	11/30/83	
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	EI (NOTE 5)	-	11/30/83	
B-38	Reconnaissance-Level Investigations	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	EI (NOTE 5)	-	11/30/83	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-45	Need for Power—Energy Conservation	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-47	Inservice Inspection of Supports—Classes 1, 2, 3, and MC Components	L. Riani	NRR/DE/MTEB	DROP	-	11/30/83	NA
B-48	BWR Control Rod Drive Mechanical Failures	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	-	11/30/83	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI (NOTE 5)	1	09/30/11	
B-50	Post-Operating Basis Earthquake Inspection	L. Riani	NRR/DE/SGEB	RI (NOTE 3)	1	06/30/85	NA
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	R. Emrit	NRR/DE/MEB	A-40	-	11/30/83	NA
B-52	Fuel Assembly Seismic and LOCA Responses	R. Emrit	NRR/DST/GIB	A-2	-	11/30/83	NA
B-53	Load Break Switch	G. Sege	NRR/DSI/PSB	RI (NOTE 3)	-	11/30/83	
B-54	Ice Condenser Containments	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	1	12/31/84	NA
B-55	Improved Reliability of Target Rock Safety Relief Valves	H. Vandermolen	NRR/DE/EMEB	NOTE 3(b)	1	06/30/00	
B-56	Diesel Reliability	W. Milstead	RES/DRPS/RPSI	NOTE 3(a)	2	06/30/95	D-19
B-57	Station Blackout	R. Emrit	NRR/DST/GIB	A-44	-	11/30/83	
B-58	Passive Mechanical Failures	L. Riani	NRR/DE/EQB	NOTE 3(b)	1	12/31/85	NA
B-59	(N-1) Loop Operation in BWRs and PWRs	L. Riani	NRR/DSI/RSB	RI (NOTE 3)	1	06/30/85	E-04,

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
							E-05
B-60	Loose Parts Monitoring Systems	R. Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	J. Pittman	RES/DST/PRAB	NOTE 3(b)	1	06/30/00	
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (NOTE 3)	-	11/30/83	NA
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	R. Emrit	NRR/DE/MEB	NOTE 3(a)	-	11/30/83	B-45
B-64	Decommissioning of Reactors	L. Riani	RES/DE/MEB	NOTE 3(a)	2	06/30/95	NA
B-65	Iodine Spiking	W. Milstead	NRR/DSI/AEB	DROP	2	12/31/84	NA
B-66	Control Room Infiltration Measurements	P. Matthews	NRR/DSI/AEB	NOTE 3(a)	-	11/30/83	
B-67	Effluent and Process Monitoring Instrumentation	L. Riani	NRR/DSI/METB	III.D.2.1	-	11/30/83	NA
B-68	Pump Overspeed during LOCA	L. Riani	NRR/DSI/ASB	DROP	-	11/30/83	NA
B-69	ECCS Leakage Ex-Containment	L. Riani	NRR/DSI/METB	III.D.1.1(1)	-	11/30/83	NA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	R. Emrit	NRR/DSI/PSB	NOTE 3(b)	-	11/30/83	
B-71	Incident Response	L. Riani	NRR	III.A.3.1	-	11/30/83	NA
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	-	NRR/DSI/RAB	LI (NOTE 5)	1	09/30/11	NA
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	D. Thatcher	NRR/DE/MEB	C-12	-	11/30/83	NA
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	W. Milstead	NRR/DE/EQB	NOTE 3(a)	-	11/30/83	
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	R. Emrit	NRR/DSI/CSB	NOTE 3(b)	-	11/30/83	NA
C-3	Insulation Usage within Containment	R. Emrit	NRR/DST/GIB	A-43	1	06/30/91	NA
C-4	Statistical Methods for ECCS Analysis	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-5	Decay Heat Update	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-6	LOCA Heat Sources	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-7	PWR System Piping	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	-	11/30/83	NA
C-8	Main Steam Line Leakage Control Systems	W. Milstead	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/90	NA
C-9	RHR Heat Exchanger Tube Failures	H. Vandermolen	NRR/DSI/RSB	DROP	-	11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	R. Emrit	NRR/DSI/AEB	NOTE 3(a)	-	11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	R. Emrit	NRR/DE/MEB	NOTE 3(b)	-	12/31/85	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
C-12	Primary System Vibration Assessment	D. Thatcher	NRR/DE/MEB	NOTE 3(b)	-	11/30/83	NA
C-13	Non-Random Failures	R. Emrit	NRR/DST/GIB	A-17	1	06/30/91	NA
C-14	Storm Surge Model for Coastal Sites	R. Emrit	NRR/DE/EHEB	LI (NOTE 3)	2	09/30/11	NA
C-15	NUREG Report for Liquid Tank Failure Analysis	-	NRR/DE/EHEB	LI (NOTE 3)	-	11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	R. Emrit	NRR/DSI/METB	NOTE 3(a)	-	11/30/83	NA
D-1	Advisability of a Seismic Scram	D. Thatcher	RES/DET/MSEB	DROP	1	12/31/98	NA
D-2	Emergency Core Cooling System Capability for Future Plants	R. Emrit	RES/DRA/ARGIB	DROP	-	12/31/88	NA
D-3	Control Rod Drop Accident	R. Emrit	NRR/DSI/CPB	NOTE 3(b)	-	11/30/83	NA
		NEW GENERIC ISSUES					
1	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	R. Emrit	NRR/DSI/METB	DROP	-	11/30/83	NA
2	Failure of Protective Devices on Essential Equipment	S. Diab	RES/DSIR/EIB	DROP	2	06/30/95	NA
3	Set Point Drift in Instrumentation	R. Emrit	NRR/DSIR/RPSIB	NOTE 3(b)	1	06/30/86	NA
4	End-of-Life and Maintenance Criteria	D. Thatcher	NRR/DE/EQB	NOTE 3(b)	-	11/30/83	NA
5	Design Check and Audit of Balance-of-Plant Equipment	J. Pittman	NRR/DSI/ASB	I.F.1	-	11/30/83	NA
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	H. Vandermolen	NRR/DSI/CPB	NOTE 3(b)	1	12/31/94	NA
7	Failures Due to Flow-Induced Vibrations	H. Vandermolen	NRR/DSI/RSB	DROP	1	06/30/91	NA
8	Inadvertent Actuation of Safety Injection in PWRs	L. Riani	NRR/DSI/RSB	I.C.1	-	11/30/83	NA
9	Reevaluation of Reactor Coolant Pump Trip Criteria	R. Emrit	NRR/DSI/RSB	II.K.3(5)	-	11/30/83	NA
10	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	R. Riggs	NRR/DSI/ICSB	DROP	-	11/30/83	NA
11	Turbine Disc Cracking	J. Pittman	NRR/DE/MTEB	A-37	-	11/30/83	NA
12	BWR Jet Pump Integrity	G. Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13	Small-Break LOCA from Extended Overheating of Pressurizer Heaters	L. Riani	NRR/DSI/RSB	DROP	-	11/30/83	NA
14	PWR Pipe Cracks	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	2	12/31/94	NA
15	Radiation Effects on Reactor Vessel Supports	R. Emrit	RES/DET/EMMEB	NOTE 3(b)	3	06/30/96	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
16	BWR Main Steam Isolation Valve Leakage Control Systems	W. Milstead	NRR/DSI/ASB	C-8	-	11/30/83	NA
17	Loss of Offsite Power Subsequent to a LOCA	L. Riani	NRR/DSI/PSB, ICSB	DROP	-	11/30/83	NA
18	Steam Line Break with Consequential Small LOCA	R. Riggs	NRR/DSI/RSB	I.C.1	-	11/30/83	NA
19	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	G. Sege	NRR/DST/GIB	A-47	-	11/30/83	NA
20	Effects of Electromagnetic Pulse on Nuclear Power Plants	D. Thatcher	NRR/DSI/ICSB	NOTE 3(b)	1	06/30/84	NA
21	Vibration Qualification of Equipment	R. Riggs	NRR/DE/EIB	DROP	2	06/30/91	NA
22	Inadvertent Boron Dilution Events	H. Vandermolen	NRR/DSI/RSB	NOTE 3(b)	2	12/31/94	NA
23	Reactor Coolant Pump Seal Failures	R. Riggs	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
24	Automatic ECCS Switchover to Recirculation	W. Milstead	RES/DET/GSIB	NOTE 3(b)	3	12/31/95	NA
25	Automatic Air Header Dump on BWR Scram System	W. Milstead	NRR/DSI/RSB	NOTE 3(a)	-	11/30/83	
26	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	R. Emrit	NRR/DSI/ASB	17	-	11/30/83	NA
27	Manual vs. Automated Actions	J. Pittman	NRR/DSI/RSB	B-17	-	11/30/83	NA
28	Pressurized Thermal Shock	R. Emrit	NRR/DST/GIB	A-49	-	11/30/83	NA
29	Bolting Degradation or Failure in Nuclear Power Plants	H. Vandermolen	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
30	Potential Generator Missiles—Generator Rotor Retaining Rings	J. Pittman	NRR/DE/MEB	DROP	1	12/31/85	NA
31	Natural Circulation Cooldown	R. Riggs	NRR/DSI/RSB	I.C.1	-	11/30/83	NA
32	Flow Blockage in Essential Equipment Caused by Corbicula	R. Emrit	NRR/DSI/ASB	51	-	11/30/83	NA
33	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	J. Pittman	NRR/DSI/ICSB	A-47	-	11/30/83	NA
34	RCS Leak	R. Riggs	NRR/DHFS/PSRB	DROP	1	06/30/84	NA
35	Degradation of Internal Appurtenances in LWRs	H. Vandermolen	NRR/DSI/CPB, RSB	DROP	2	12/31/98	NA
36	Loss of Service Water	L. Riani	NRR/DSI/ASB, AEB, RSB	NOTE 3(b)	3	06/30/91	NA
37	Steam Generator Overfill and Combined Primary and Secondary Blowdown	L. Riani	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C.1(2)	1	06/30/85	NA
38	Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris	R. Emrit	RES/DSIR/RPSIB	DROP	2	06/30/95	NA
39	Potential for Unacceptable Interaction between the CRD	J. Pittman	NRR/DSI/ASB	25	1	06/30/95	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	System and Non-Essential Control Air System						
40	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	L. Riani	NRR/DSI/ASB	NOTE 3(a)	1	06/30/84	B-65
41	BWR Scram Discharge Volume Systems	H. Vandermolen	NRR/DSI/RSB	NOTE 3(a)	-	11/30/83	B-58
42	Combination Primary/Secondary System LOCA	R. Riggs	NRR/DSI/RSB	I.C.1	1	06/30/85	NA
43	Reliability of Air Systems	W. Milstead	RES/DSIR/RPSI	NOTE 3(a)	2	12/31/88	B-107
44	Failure of Saltwater Cooling System	W. Milstead	NRR/DSI/ASB	43	1	12/31/88	NA
45	Inoperability of Instrumentation Due to Extreme Cold Weather	W. Milstead	NRR/DSI/ICSB	NOTE 3(a)	2	06/30/91	
46	Loss of 125 Volt DC Bus	G. Sege	NRR/DSI/PSB	76	-	11/30/83	NA
47	Loss of Offsite Power	D. Thatcher	NRR/DSI/RSB, ASB	NOTE 3(b)	-	11/30/83	
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors	G. Sege	NRR/DSI/PSB	128	1	12/31/86	NA
49	Interlocks and LCOs for Redundant Class 1E Tie-Breakers	G. Sege	NRR/DSI/PSB	128	3	06/30/91	NA
50	Reactor Vessel Level Instrumentation in BWRs	D. Thatcher	NRR/DSI/RSB, ICSB	NOTE 3(b)	1	12/31/84	NA
51	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	R. Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	L-913
52	SSW Flow Blockage by Blue Mussels	R. Emrit	NRR/DSI/ASB	51	-	11/30/83	NA
53	Consequences of a Postulated Flow Blockage Incident in a BWR	H. Vandermolen	NRR/DSI/CPB, RSB	DROP	1	12/31/84	NA
54	Valve Operator-Related Events Occurring during 1978, 1979, and 1980	L. Riani	NRR/DE/MEB	II.E.6.1	1	06/30/85	NA
55	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	R. Emrit	NRR/DSI/PSB	DROP	2	06/30/91	NA
56	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	L. Riani	NRR/DHFS/HFEB	A-47, I.D.1	-	11/30/83	NA
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	3	06/30/95	NA
58	Inadvertent Containment Flooding	G. Sege	NRR/DSI/ASB, CSB	DROP	-	11/30/83	
59	Technical Specification Requirements for Plant Shutdown When Equipment for Safe Shutdown is Degraded or Inoperable	R. Emrit	NRR/DST/TSIP	RI (NOTE 5)	2	09/30/11	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
60	Lamellar Tearing of Reactor Systems Structural Supports	L. Riani	NRR/DST/GIB	A-12	-	11/30/83	NA
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/86	NA
62	Reactor Systems Bolting Applications	R. Riggs	RES/DSIR/EIB	29	1	12/31/88	NA
63	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	J. Pittman	RES/DRA/ARGIB	DROP	1	06/30/90	NA
64	Identification of Protection System Instrument Sensing Lines	D. Thatcher	NRR/DSI/ICSB	NOTE 3(b)	-	11/30/83	
65	Probability of Core-Melt Due to Component Cooling Water System Failures	H. Vandermolen	NRR/DSI/ASB	23	1	12/31/86	NA
66	Steam Generator Requirements	R. Riggs	NRR/DEST/EMTB	NOTE 3(b)	2	12/31/88	NA
<u>67</u>	<u>Steam Generator Staff Actions</u>						
67.2.1	Integrity of Steam Generator Tube Sleeves	R. Riggs	NRR/DE/MEB	135	5	09/30/11	NA
67.3.1	Steam Generator Overfill	R. Riggs	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C.1	5	09/30/11	NA
67.3.2	Pressurized Thermal Shock	R. Riggs	NRR/DST/GIB	A-49	5	09/30/11	NA
67.3.3	Improved Accident Monitoring	R. Riggs	NRR/DSI/ICSB	NOTE 3(a)	5	09/30/11	A-17
67.3.4	Reactor Vessel Inventory Measurement	R. Riggs	NRR/DSI/CPB	II.F.2	5	09/30/11	NA
67.4.1	RCP Trip	R. Riggs	NRR/DSI/RSB	II.K.3(5)	5	09/30/11	G-01
67.4.2	Control Room Design Review	R. Riggs	NRR/DHFS/HFEB	I.D.1	5	09/30/11	F-08
67.4.3	Emergency Operating Procedures	R. Riggs	NRC/DHFS/PSRB	I.C.1	5	09/30/11	F-05
67.5.1	Reassessment of Radiological Consequences	R. Riggs	RES/DRPS/RPSI	LI (NOTE 3)	5	09/30/11	NA
67.5.2	Reevaluation of SGTR Design Basis	R. Riggs	RES/DRPS/RPSI	LI (67.5.1)	5	09/30/11	NA
67.5.3	Secondary System Isolation	R. Riggs	NRR/DSI/RSB	DROP	5	09/30/11	NA
67.6.0	Organizational Responses	R. Riggs	OIE/DEPER/IRDB	III.A.3	5	09/30/11	NA
67.7.0	Improved Eddy Current Tests	R. Riggs	RES/DE/EIB	135	5	09/30/11	NA
67.8.0	Denting Criteria	R. Riggs	NRR/DE/MTEB	135	5	09/30/11	NA
67.9.0	Reactor Coolant System Pressure Control	R. Riggs	NRR/DSI/GIB, NRR/DSI/RSB	A-45, I.C.1 (2,3)	5	09/30/11	NA
67.10.0	Supplemental Tube Inspections	R. Riggs	NRR/DL/ORAB	LI (NOTE 5)	5	09/30/11	NA
68	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	J. Pittman	NRR/DSI/ASB	124	3	06/30/91	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
69	Make-up Nozzle Cracking in B&W Plants	R. Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	B43
70	PORV and Block Valve Reliability	R. Riggs	RES/DE/EIB	NOTE 3(a)	3	06/30/91	
71	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	J. Pittman	RES/DRA/ARGIB	DROP	3	06/30/01	NA
72	Control Rod Drive Guide Tube Support Pin Failures	R. Riggs	RES	DROP	1	06/30/91	NA
73	Detached Thermal Sleeves	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	3	06/30/95	NA
74	Reactor Coolant Activity Limits for Operating Reactors	W. Milstead	NRR/DSI/AEB	DROP	1	06/30/86	NA
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	R. Emrit	RES/DRA/ARGIB	NOTE 3(a)	1	06/30/90	B-76, B-77, B-78, B-79, B-80, B-81, B-82, B-85 B-86, B-87, B-88, B-89, B-90, B-91, B-92, B-93
76	Instrumentation and Control Power Interactions	R. Zimmerman	RES/DSIR/EIB	DROP	3	06/30/95	NA
77	Flooding of Safety Equipment Compartments by Backflow Through Floor Drains	L. Riani	RES/DE/EIB	A-17	-	12/31/87	NA
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	C. Rourk	RES/DET/GSIB	NOTE 3(b)	3	12/31/97	
79	Unanalyzed Reactor Vessel Thermal Stress during Natural Convection Cooldown	L. Riani	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
80	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	4	06/30/06	NA
81	Impact of Locked Doors and Barriers on Plant and Personnel Safety	C. Rourk	RES/DSIR/EIB	DROP	5	09/30/11	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
82	Beyond Design-Basis Accidents in Spent Fuel Pools	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	3	06/30/04	NA
83	Control Room Habitability	R. Emrit	RES/DST/AEB	NOTE 3(b)	3	06/30/03	NA
84	CE PORVs	R. Riggs	RES/DSIR/RPSI	NOTE 3(b)	2	06/30/90	NA
85	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	W. Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA
86	Long-Range Plan for Dealing with Stress-Corrosion Cracking in BWR Piping	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87	Failure of HPCI Steam Line without Isolation	J. Pittman	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
88	Earthquakes and Emergency Planning	R. Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89	Stiff Pipe Clamps	T.Y. Chang	RES/DSIR/EIB	MEDIUM	2	06/30/95	NA
90	Technical Specifications for Anticipatory Trips	H. Vandermolen	NRR/DSI/RSB, ICSB	DROP	2	12/31/98	NA
91	Main Crankshaft Failures in Transamerica Delaval Emergency Diesel Generators	R. Emrit	RES/DRA/ARGIB	NOTE 3(b)	-	12/31/87	NA
92	Fuel Crumbling during LOCA	H. Vandermolen	NRR/DSI/RSB, CPB	DROP	1	12/31/98	NA
93	Steam Binding of Auxiliary Feedwater Pumps	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)	-	06/30/88	B-98
94	Additional Low Temperature Overpressure Protection for Light-Water Reactors	J. Pittman	RES/DSIR/RPSI	NOTE 3(a)	-	06/30/90	
95	Loss of Effective Volume for Containment Recirculation Spray	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	-	06/30/90	NA
96	RHR Suction Valve Testing	W. Milstead	RES/DRA/ARGIB	105	-	06/30/90	NA
97	PWR Reactor Cavity Uncontrolled Exposures	H. Vandermolen	NRR/DSI/RAB	III.D.3.1	-	06/30/85	NA
98	CRD Accumulator Check Valve Leakage	J. Pittman	NRR/DSI/ASB	DROP	-	06/30/85	NA
99	RCS/RHR Suction Line Valve Interlock on PWRs	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)	3	06/30/91	L-817
100	Once-Through Steam Generator Level	J. Jackson	RES/DSIR/EIB	DROP	1	06/30/95	NA
101	BWR Water Level Redundancy	H. Vandermolen	RES/DE/EIB	NOTE 3(b)	1	06/30/89	NA
102	Human Error in Events Involving Wrong Unit or Wrong Train	R. Emrit	NRR/DLPQ/LPEB	NOTE 3(b)	2	12/31/88	NA
103	Design for Probable Maximum Precipitation	R. Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	NA
104	Reduction of Boron Dilution Requirements	J. Pittman	RES/DRA/ARGIB	DROP	-	12/31/88	NA
105	Interfacing Systems LOCA at LWRs	W. Milstead	RES/DE/EIB	NOTE 3(b)	4	06/30/95	NA
106	Piping and Use of Highly Combustible Gases in Vital Areas	W. Milstead	RES/DRPS	NOTE 3(b)	2	06/30/95	NA
107	Main Transformer Failures	W. Milstead	RES/DRA/ARGIB	DROP	3	06/30/00	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
108	BWR Suppression Pool Temperature Limits	L. Riani	NRR/DSI/CSB	RI (NOTE 3)	-	06/30/85	NA
109	Reactor Vessel Closure Failure	R. Riggs	RES/DRA/ARGIB	DROP	-	06/30/90	NA
110	Equipment Protective Devices on Engineered Safety Features	S. Diab	RES/DSIR/EIB	DROP	1	06/30/95	NA
111	Stress-Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	R. Riggs	NRR/DE/MTEB	LI (NOTE 5)	2	09/30/11	NA
112	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	J. Pittman	NRR/DSI/ICSB	RI (NOTE 3)	-	12/31/85	NA
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	R. Riggs	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
114	Seismic-Induced Relay Chatter	R. Riggs	NRR/DSRO/SPEB	A-46	1	06/30/91	NA
115	Enhancement of the Reliability of Westinghouse Solid State Protection System	W. Milstead	RES/DRPS/RPSI	NOTE 3(b)	2	06/30/00	NA
116	Accident Management	J. Pittman	RES/DRA/ARGIB	S	-	06/30/91	NA
117	Allowable Time for Diverse Simultaneous Equipment Outages	J. Pittman	RES/DRA/ARGIB	DROP	-	06/30/90	NA
118	Tendon Anchorage Failure	S. Shaukat	RES/DSIR/EIB	NOTE 3(a)	1	06/30/95	NA
119	Piping Review Committee Recommendations						
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	R. Riggs	NRR/DE	RI (NOTE 3)	4	09/30/11	NA
119.2	Piping Damping Values	R. Riggs	NRR/DE	RI (DROP)	4	09/30/11	NA
119.3	Decoupling the OBE from the SSE	R. Riggs	NRR/DE	RI (S)	4	09/30/11	NA
119.4	BWR Piping Materials	R. Riggs	NRR/DE	RI (NOTE 5)	4	09/30/11	NA
119.5	Leak Detection Requirements	R. Riggs	NRR/DE	RI (NOTE 5)	4	09/30/11	NA
120	On-Line Testability of Protection Systems	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
121	Hydrogen Control for Large, Dry PWR Containments	R. Emrit	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
122	Davis-Besse Loss of All Feedwater Event of June 9, 1985: Short-Term Actions						
122.1	Potential Inability to Remove Reactor Decay Heat						
122.1.a	Failure of Isolation Valves in Closed Position	H. Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA
122.1.b	Recovery of Auxiliary Feedwater	H. Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
122.1.c.	Interruption of Auxiliary Feedwater Flow	H. Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA
122.2	Initiating Feed-and-Bleed	H. Vandermolen	NRR/DEST/SRXB	NOTE 3(b)	4	12/31/98	NA
122.3	Physical Security System Constraints	H. Vandermolen	NRR/DSRO/SPEB	DROP	4	12/31/98	NA
123	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	W. Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
124	Auxiliary Feedwater System Reliability	R. Emrit	NRR/DEST/SRXB	NOTE 3(a)	3	06/30/91	
<u>125</u>	<u>Davis-Besse Loss of All Feedwater Event of June 9, 1985: Long-Term Actions</u>						
125.I.1	Availability of the Shift Technical Advisor	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.2	PORV Reliability						
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	H. Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	H. Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.c	Need for Additional Protection against PORV Failure	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	H. Vandermolen	NRR/DSRO/SPEB	A-45	7	12/31/98	NA
125.I.3	SPDS Availability	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	7	12/31/98	NA
125.I.4	Plant-Specific Simulator	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.5	Safety Systems Tested in All Conditions Required by DBA	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.6	Valve Torque Limit and Bypass Switch Settings	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7	Operator Training Adequacy						
125.I.7.a	Recover Failed Equipment	J. Pittman	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7.b	Realistic Hands-On Training	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.1	Need for Additional Actions on AFW Systems						
125.II.1.a	Two-Train AFW Unavailability	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.1.b	Review Existing AFW Systems for Single Failure	H. Vandermolen	NRR/DSRO/SPEB	124	7	12/31/98	NA
125.II.1.c	NUREG-0737 Reliability Improvements	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Interactions in B&W Plants						
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.4	Thermal Stress of OTSG Components	R. Riggs	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	
125.II.6	Reexamine PRA Estimates of Core Damage Risk from Loss of All Feedwater	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator during a Line Break	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	7	12/31/98	NA
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.9	Enhanced Feed-and-Bleed Capability	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.10	Hierarchy of Impromptu Operator Actions	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.12	Adequacy of Training Regarding PORV Operation	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.13	Operator Job Aids	J. Pittman	NRR/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
126	Reliability of PWR Main Steam Safety Valves	R. Riggs	RES/DRA/ARGIB	LI (NOTE 3)	-	06/30/88	NA
127	Maintenance and Testing of Manual Valves in Safety-Related Systems	J. Pittman	RES/DRA/ARGIB	DROP	1	09/30/11	NA
128	Electrical Power Reliability	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
129	Valve Interlocks to Prevent Vessel Drainage during Shutdown Cooling	W. Milstead	RES/DRA/ARGIB	DROP	-	06/30/90	NA
130	Essential Service Water Pump Failures at Multiplant Sites	R. Riggs	RES/DSIR/RPSIB	NOTE 3(a)	2	12/31/95	
131	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse-Designed Plants	R. Riggs	RES/DRA/ARGIB	S	1	06/30/91	NA
132	RHR System Inside Containment	N. Su	RES/DSIR/SAIB	DROP	1	12/31/95	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
133	Update Policy Statement on Nuclear Plant Staff Working Hours	J. Pittman	NRR/DLPQ/LHFB	LI (NOTE 3)	1	12/31/91	NA
134	Rule on Degree and Experience Requirement	J. Pittman	RES/DRA/RDB	NOTE 3(b)	-	12/31/89	NA
135	Steam Generator and Steam Line Overfill	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
136	Storage and Use of Large Quantities of Cryogenic Combustibles Onsite	W. Milstead	RES/DRA/ARGIB	LI (NOTE 3)	-	06/30/88	NA
137	Refueling Cavity Seal Failure	W. Milstead	RES/DRA/ARGIB	DROP	-	06/30/90	NA
138	Deinerting of BWR Mark I and II Containments during Power Operations upon Discovery of RCS Leakage or a Train of a Safety System Inoperable	W. Milstead	RES/DSIR/SAIB	DROP	2	12/31/98	NA
139	Thinning of Carbon Steel Piping in LWRs	R. Riggs	RES/DRA/ARGIB	RI (NOTE 3)	1	06/30/95	NA
140	Fission Product Removal Systems	R. Riggs	RES/DRA/ARGIB	DROP	-	06/30/90	NA
141	Large-Break LOCA with Consequential SGTR	R. Riggs	RES/DRA/ARGIB	DROP	-	06/30/90	NA
142	Leakage through Electrical Isolators in Instrumentation Circuits	W. Milstead	RES/DSIR/EIB	NOTE 3(b)	4	12/31/97	NA
143	Availability of Chilled Water Systems and Room Cooling	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
144	Scram without a Turbine/Generator Trip	C. Hrabal	RES/DSIR/EIB	DROP	2	12/31/98	NA
145	Actions to Reduce Common-Cause Failures	D. Rasmuson	RES/DST/PRAB	NOTE 3(b)	3	06/30/00	NA
146	Support Flexibility of Equipment and Components	T. Y. Chang	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
147	Fire-Induced Alternate Shutdown/Control Room Panel Interactions	W. Milstead	RES/DSIR/SAIB	LI (NOTE 3)	1	06/30/94	NA
148	Smoke Control and Manual Fire-Fighting Effectiveness	D. Basdekas	RES/DSIR/RPSIB	LI (NOTE 3)	1	06/30/00	NA
149	Adequacy of Fire Barriers	R. Emrit	RES/DSIR/EIB	DROP	2	12/31/98	NA
150	Overpressurization of Containment Penetrations	W. Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
151	Reliability of Anticipated Transient without SCRAM Recirculation Pump Trip in BWRs	W. Milstead	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
152	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	R. Emrit	RES/DSIR/EIB	DROP	3	06/30/01	NA
153	Loss of Essential Service Water in LWRs	R. Riggs	RES/DRA/ARGIB	NOTE 3(b)	2	12/31/95	NA
154	Adequacy of Emergency and Essential Lighting	R. Woods	RES/DSIR/SAIB	DROP	2	12/31/98	NA
155	Generic Concerns Arising from TMI-2 Cleanup						
155.1	More Realistic Source Term Assumptions	R. Emrit	RES/DST/AEB	NOTE 3(a)	3	09/30/11	NA
155.2	Establish Licensing Requirements for Non-Operating Facilities	R. Emrit	RES/DSIR/EIB	RI (NOTE 5)	3	09/30/11	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
155.3	Improve Design Requirements for Nuclear Facilities	R. Emrit	RES/DSIR/EIB	DROP	3	09/30/11	NA
155.4	Improve Criticality Calculations	R. Emrit	RES/DSIR/EIB	DROP	3	09/30/11	NA
155.5	More Realistic Severe Reactor Accident Scenario	R. Emrit	RES/DSIR/EIB	DROP	3	09/30/11	NA
155.6	Improve Decontamination Regulations	R. Emrit	RES/DSIR/EIB	DROP	3	09/30/11	NA
155.7	Improve Decommissioning Regulations	R. Emrit	RES/DSIR/EIB	DROP	3	09/30/11	NA
156	Systematic Evaluation Program						
156.1.1	Settlement of Foundations and Buried Equipment	T.Y. Chang	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.1.2	Dam Integrity and Site Flooding	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.3	Site Hydrology and Ability to Withstand Floods	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.4	Industrial Hazards	C. Ferrell	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.5	Tornado Missiles	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.6	Turbine Missiles	R. Emrit	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.2.1	Severe Weather Effects on Structures	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.2.2	Design Codes, Criteria, and Load Combinations	R. Kirkwood	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.2.3	Containment Design and Inspection	S. Shaukat	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.2.4	Seismic Design of Structures, Systems, and Components	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.1.1	Shutdown Systems	R. Woods	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.1.2	Electrical Instrumentation and Controls	R. Woods	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.2	Service and Cooling Water Systems	N. Su	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.3	Ventilation Systems	G. Burdick	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.4	Isolation of High- and Low-Pressure Systems	G. Burdick	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.5	Automatic ECCS Switchover	W. Milstead	RES/DSIR/SAIB	24	8	06/30/08	NA
156.3.6.1	Emergency AC Power	R. Emrit	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.3.6.2	Emergency DC Power	C. Rourk	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.3.8	Shared Systems	R. Emrit	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.4.1	RPS and ESFS Isolation	R. Emrit	RES/DSIR/EIB	142	8	06/30/08	NA
156.4.2	Testing of the RPS and ESFS	T.Y. Chang	RES/DSIR/SAIB	120	8	06/30/08	NA
156.6.1	Pipe Break Effects on Systems and Components	H. Vandermolen	RES/DRA/OEGIB	NOTE 3(b)	8	06/30/08	NA
157	Containment Performance	J. Shaperow	RES/DSIR/SAIB	NOTE 3(b)	-	06/30/95	NA
158	Performance of Power-Operated Valves Under Design Basis Conditions	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
159	Qualification of Safety-Related Pumps While Running on Minimum Flow	N. Su	RES/DSIR/SAIB	DROP	1	06/30/95	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
160	Spurious Actions of Instrumentation Upon Restoration of Power	C. Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
161	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	C. Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
162	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	U. Cheh	RES/DSIR/SAIB	DROP	1	06/30/95	NA
163	Multiple Steam Generator Tube Leakage	E. Murphy	NRR/DCI/CSG	NOTE 3(b)	2	06/30/10	
164	Neutron Fluence in Reactor Vessel	R. Emrit	RES/DSIR/EIB	DROP	1	06/30/95	NA
165	Safety and Safety/Relief Valve Reliability	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
166	Adequacy of Fatigue Life of Metal Components	R. Emrit	NRR/DE/EMEB	NOTE 3(b)	2	12/31/97	NA
167	Hydrogen Storage Facility Separation	G. Burdick	RES/DSIR/SAIB	DROP	2	09/30/11	NA
168	Environmental Qualification of Electrical Equipment	R. Emrit	NRR/DSSA/SPLB	NOTE 3(b)	3	06/30/04	NA
169	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	R. Emrit	RES/DET/GSIB	DROP	1	06/30/00	NA
170	Fuel Damage Criteria for High Burnup Fuel	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/01	NA
171	ESF Failure from LOOP Subsequent to a LOCA	C. Rourk	RES/DET/GSIB	NOTE 3(b)	1	12/31/98	NA
172	Multiple System Responses Program	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/02	NA
173	Spent Fuel Storage Pool						
173.A	Operating Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
173.B	Permanently Shutdown Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
174	Fastener Gaging Practices						
174.A	SONGS Employees' Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
174.B	Johnson Gage Company Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
175	Nuclear Power Plant Shift Staffing	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
176	Loss of Fill-Oil in Rosemount Transmitters	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
177	Vehicle Intrusion at TMI	R. Emrit	RES/DET/GSIB	NOTE 3(a)	1	06/30/00	NA
178	Effect of Hurricane Andrew on Turkey Point	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	2	06/30/00	
179	Core Performance	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	2	09/30/11	
180	Notice of Enforcement Discretion	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	1	06/30/00	
181	Fire Protection	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	2	09/30/11	
182	General Electric Extended Power Uprate	R. Emrit	RES/DET/GSIB	RI (NOTE 5)	2	09/30/11	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
183	Cycle-Specific Parameter Limits in Technical Specifications	R. Emrit	RES/DET/GSIB	RI (NOTE 3)	2	06/30/00	
184	Endangered Species	R. Emrit	RES/DET/GSIB	EI (NOTE 5)	2	09/30/11	
185	Control of Recriticality Following Small-Break LOCA in PWRs	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
186	Potential Risk and Consequences of Heavy Load Drops	S. Jones	NRR/DSS/SBP	ACTIVE	-	06/30/04	
187	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants	H. Vandermolen	RES/DSARE/REAHFB	DROP	-	06/30/01	NA
188	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion during a Severe Accident	S. Jones	NRR/DSS/SBP	ROI	1	06/30/08	
190	Fatigue Evaluation of Metal Components for 60-Year Plant Life	S. Shaukat	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
191	Assessment of Debris Accumulation on PWR Sump Performance	S. Bailey	NRR/DSS/SSIB	ROI	2	06/30/08	
192	Secondary Containment Drawdown Time	H. Vandermolen	RES/DSARE/REAHFB	DROP	-	06/30/03	NA
193	BWR ECCS Suction Concerns	J. Lane	RES/DRA/OEGIB	ACTIVE	-	06/30/04	
194	Implications of Updated Probabilistic Seismic Hazard Estimates	D. Harrison	NRR/DSSA/SPSB	DROP	-	06/30/04	NA
195	Hydrogen Combustion in Foreign BWR Piping	H. Vandermolen	RES/DSARE/REAHFB	DROP	-	06/30/04	NA
196	Boral Degradation	H. Vandermolen	RES/DSARE/ARREB	NOTE 3(b)	1	06/30/07	NA
197	Iodine Spiking Phenomena	H. Vandermolen	RES/DSARE/ARREB	DROP	-	06/30/06	NA
198	Hydrogen Combustion in PWR Piping	H. Vandermolen	RES/DRASP/OERA	DROP	-	06/30/07	NA
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	K. Manoly	NRR/DE	ROI	1	09/30/11	
200	Tin Whiskers	C. Antonescu	RES/DRASP/OERA	DROP	-	06/30/07	NA
201	Small-Break LOCA and Loss of Offsite Power Scenario	A. Salomon	RES/DRASP/OERA	DROP	-	06/30/07	NA
202	Spent Fuel Pool Leakage Limits	T. Mitts	RES/DRASP/OERA	DROP	-	06/30/07	NA
203	Potential Safety Issues with Cranes that Lift Spent Fuel Casks	T. Mitts	RES/DRASP/OERA	DROP	-	06/30/07	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	J. Pittman	RES/DRPS/RHFB	NOTE 3(a)	2	06/30/89	
HF1.2	Engineering Expertise on Shift	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA
HF1.3	Guidance on Limits and Conditions of Shift Work	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA
HF2	TRAINING						
HF2.1	Evaluate Industry Training	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	2	09/30/11	NA
HF2.2	Evaluate INPO Accreditation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	2	09/30/11	NA
HF2.3	Revise SRP Section 13.2	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	2	09/30/11	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.2	Develop License Examination Handbook	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	J. Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	J. Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	J. Pittman	NRR/DLPQ/LHFB	NOTE 3(b)	7	09/30/11	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	7	09/30/11	NA
HF4.3	Criteria for Safety-Related Operator Actions	J. Pittman	NRR/DHFT/HFIB	B-17	7	09/30/11	NA
HF4.4	Guidelines for Upgrading Other Procedures	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	7	09/30/11	NA
HF4.5	Application of Automation and Artificial Intelligence	J. Pittman	NRR/DHFT/HFIB	HF5.2	7	09/30/11	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.3	Evaluation of Operational Aid Systems	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF5.4	Computers and Computer Displays	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						
HF7.1	Human Error Data Acquisition	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	2	09/30/11	NA
HF7.2	Human Error Data Storage and Retrieval	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	2	09/30/11	NA
HF7.3	Reliability Evaluation Specialist Aids	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	2	09/30/11	NA
HF7.4	Safety Event Analysis Results Applications	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	2	09/30/11	NA
HF8	Maintenance and Surveillance Program	J. Pittman	NRR/DLPQ/LPEB	NOTE 3(b)	2	06/30/88	NA
	CHERNOBYL ISSUES						
CH1	ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES						
CH1.1	Administrative Controls To Ensure That Procedures Are Followed and That Procedures Are Adequate						
CH1.1A	Symptom-Based EOPs	R. Emrit	NRR/DLPQ/LHFB	LI (NOTE 5)	1	09/30/11	NA
CH1.1B	Procedure Violations	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)	1	09/30/11	NA
CH1.2	Approval of Tests and Other Unusual Operations						
CH1.2A	Test, Change, and Experiment Review Guidelines	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)	1	09/30/11	NA
CH1.2B	NRC Testing Requirements	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)	1	09/30/11	NA
CH1.3	Bypassing Safety Systems						

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH1.3A	Revise Regulatory Guide 1.47	R. Emrit	RES/DE/EMEB	LI (NOTE 5)	1	09/30/11	NA
CH1.4	Availability of Engineered Safety Features						
CH1.4A	Engineered Safety Feature Availability	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)	1	09/30/11	NA
CH1.4B	Technical Specifications Bases	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)	1	09/30/11	NA
CH1.4C	Low Power and Shutdown	R. Emrit	RES/DSR/PRAB	LI (NOTE 5)	1	09/30/11	NA
CH1.5	Operating Staff Attitudes Toward Safety	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)	1	09/30/11	NA
CH1.6	Management Systems						
CH1.6A	Assessment of NRC Requirements on Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)	1	09/30/11	NA
CH1.7	Accident Management						
CH1.7A	Accident Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)	1	09/30/11	NA
CH2	DESIGN						
CH2.1	Reactivity Accidents						
CH2.1A	Reactivity Transients	R. Emrit	RES/DSR/RPSB	LI (NOTE 5)	1	09/30/11	NA
CH2.2	Accidents at Low Power and at Zero Power	R. Emrit	RES/DRA/ARGIB	CH1.4	1	09/30/11	NA
CH2.3	Multiple-Unit Protection						
CH2.3A	Control Room Habitability	R. Emrit	RES/DRA/ARGIB	83	1	09/30/11	NA
CH2.3B	Contamination Outside Control Room	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)	1	09/30/11	NA
CH2.3C	Smoke Control	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)	1	09/30/11	NA
CH2.3D	Shared Shutdown Systems	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)	1	09/30/11	NA
CH2.4	Fire Protection						
CH2.4A	Firefighting with Radiation Present	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)	1	09/30/11	NA
CH3	CONTAINMENT						
CH3.1	Containment Performance during Severe Accidents						
CH3.1A	Containment Performance	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)	1	09/30/11	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH3.2	Filtered Venting						
CH3.2A	Filtered Venting	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)	1	09/30/11	NA
CH4	EMERGENCY PLANNING						
CH4.1	Size of the Emergency Planning Zones	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)	1	09/30/11	NA
CH4.2	Medical Services	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)	1	09/30/11	NA
CH4.3	Ingestion Pathway Measures						
CH4.3A	Ingestion Pathway Protective Measures	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)	1	09/30/11	NA
CH4.4	Decontamination and Relocation						
CH4.4A	Decontamination	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)	1	09/30/11	NA
CH4.4B	Relocation	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)	1	09/30/11	NA
CH5	SEVERE ACCIDENT PHENOMENA						
CH5.1	Source Term						
CH5.1A	Mechanical Dispersal in Fission Product Release	R. Emrit	RES/DSR/AEB	LI (NOTE 5)	1	09/30/11	NA
CH5.1B	Stripping in Fission Product Release	R. Emrit	RES/DSR/AEB	LI (NOTE 5)	1	09/30/11	NA
CH5.2	Steam Explosions						
CH5.2A	Steam Explosions	R. Emrit	RES/DSR/AEB	LI (NOTE 5)	1	09/30/11	NA
CH5.3	Combustible Gas	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)	1	09/30/11	NA
CH6	GRAPHITE-MODERATED REACTORS						
CH6.1	Graphite-Moderated Reactors						
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.1B	Structural Graphite Experiments	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.2	Assessment	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA

TABLE III

SUMMARY OF THE STATUS OF ALL GENERIC SAFETY ISSUES

Legend

ACTIVE	Generic issue that involves actions under the GIP
DROP	Issue dropped from further pursuit as a generic issue
EI	Environmental issue
GSI	Generic safety issue
LI	Licensing issue
MEDIUM	Medium safety priority
NOTE 5	Issue that is not a generic safety issue but should be assigned resources for completion. As clarified by SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011, Generic Issues Program will not pursue any further actions toward resolution of Licensing and Regulatory impact issues.
RI	Regulatory impact issue
ROI	Regulatory office implementation: A formal GI for which RES actions of safety/risk assessment or regulatory assessment are complete and remaining actions reside with program offices (e.g., regulatory compliance, reactor oversight process, rulemaking, further research, coordination with industry initiatives)
USI	Unresolved safety issue

<table>
<tr><th rowspan="2">Action Item / Issue Group</th><th colspan="2">Legacy Program</th><th rowspan="2">ACTIVE</th><th rowspan="2">ROI</th><th colspan="6">Actions Completed</th><th rowspan="2">Total</th></tr>
<tr><th>NOTE 5</th><th>MEDIUM</th><th>Resolved with Regulatory Product</th><th>Resolved without Regulatory Product</th><th>DROP</th><th>NOTE 5</th><th>Resolution by NUREG-0737</th><th>Covered by Other Issues</th></tr>
<tr><td colspan="12">TMI ACTION PLAN ITEM (369)</td></tr>
<tr><td>GSI</td><td>-</td><td>-</td><td>-</td><td>-</td><td>66</td><td>69</td><td>21</td><td>-</td><td>84</td><td>46</td><td>286</td></tr>
<tr><td>LI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">75</td><td>-</td><td>8</td><td>-</td><td>-</td><td>83</td></tr>
<tr><td colspan="12">TASK ACTION PLAN ITEMS (142)</td></tr>
<tr><td>USI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">27</td><td>-</td><td>-</td><td>-</td><td>-</td><td>27</td></tr>
<tr><td>GSI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">36</td><td>14</td><td>-</td><td>-</td><td>20</td><td>70</td></tr>
<tr><td>RI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">6</td><td>-</td><td>1</td><td>-</td><td>-</td><td>7</td></tr>
<tr><td>LI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">11</td><td>-</td><td>12</td><td>-</td><td>-</td><td>23</td></tr>
<tr><td>EI</td><td>2</td><td>-</td><td>-</td><td>-</td><td colspan="2">13</td><td>-</td><td>-</td><td>-</td><td>-</td><td>15</td></tr>
<tr><td colspan="12">NEW GENERIC ISSUES (283)</td></tr>
<tr><td>GSI</td><td>-</td><td>1</td><td>2</td><td>3</td><td>23</td><td>66</td><td>108</td><td>-</td><td>-</td><td>54</td><td>257</td></tr>
<tr><td>RI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">5</td><td>1</td><td>5</td><td>-</td><td>1</td><td>12</td></tr>
<tr><td>LI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">8</td><td>-</td><td>4</td><td>-</td><td>1</td><td>13</td></tr>
<tr><td>EI</td><td>1</td><td>-</td><td>-</td><td>-</td><td colspan="2">-</td><td>-</td><td>-</td><td>-</td><td>-</td><td>1</td></tr>
<tr><td colspan="12">HUMAN FACTORS ISSUES (27)</td></tr>
<tr><td>GSI</td><td>-</td><td>-</td><td>-</td><td>-</td><td>1</td><td>7</td><td>-</td><td>-</td><td>-</td><td>8</td><td>16</td></tr>
<tr><td>LI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">3</td><td>-</td><td>8</td><td>-</td><td>-</td><td>11</td></tr>
<tr><td colspan="12">CHERNOBYL ISSUES (32)</td></tr>
<tr><td>LI</td><td>-</td><td>-</td><td>-</td><td>-</td><td colspan="2">7</td><td>-</td><td>23</td><td>-</td><td>2</td><td>32</td></tr>
<tr><td>TOTAL:</td><td>3</td><td>1</td><td>2</td><td>3</td><td colspan="2">423</td><td>144</td><td>61</td><td>84</td><td>132</td><td>853</td></tr>
</table>

TASK I.F: QUALITY ASSURANCE

The objective of this task was to improve the quality assurance program (QA) for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities were conducted in a manner commensurate with their importance to safety.

ITEM I.F.1: EXPAND QA LIST

Description

Historical Background

The Three Mile Island (TMI) Action Plan[48] identified that several systems important to the safety of Three Mile Island Unit 2 (TMI-2) were not designed, fabricated, and maintained at a level equivalent to their safety importance; i.e., they were not on the QA list for the plant. This condition existed at other plants and resulted primarily from the lack of clarity in U.S. Nuclear Regulatory Commission (NRC) guidance on graded protection. Evaluation of this issue included the consideration of Issue 5 (see Section 3).

Safety Significance

One of the difficulties in establishing a QA list based on safety importance was the absence of relative risk assignments to equipment. At the time this issue was initially evaluated, QA requirements were applied principally to structures, systems, and components that prevented or mitigated the consequences of postulated accidents that could cause undue risk to the health and safety of the public (Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”).

Possible Solution

The TMI Action Plan stated that the NRC would develop guidance for licensees to expand their QA lists to cover equipment important to safety (ITS) and rank the equipment in order of its importance to safety. Experience in the use of the revised Office of Nuclear Reactor Regulation review procedure for developing QA lists for individual operating license applicants was to be factored into the generic guidance to be developed and when determining backfit requirements.[48] At the time this issue was identified, there was a task underway to define the applicability of Appendix B to 10 CFR Part 50 to equipment that met the requirements of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50.

Priority Determination

The principal benefit to be derived from an expanded QA list was the knowledge that adequate guidance provided to each licensee to establish QA programs and requirements that were commensurate with the safety importance of structures, systems, and components, as determined from completed risk assessments. This guidance would not only result in the inclusion or addition to each licensee's QA list of other ITS systems that were previously excluded but would also aid in clarifying the QA level of effort deemed necessary.

The risk reduction was probably proportionate to the difference between what would normally be the level of effort expended and the level defined. At the time this issue was initially evaluated, there was no measure of risk variation as a function of the variance in QA level of effort. However, it appeared reasonable to assume that a significant reduction in public risk could be achieved at those plants where the QA levels were held to the previous minimum acceptable level. Important questions to which there were no answers were (1) the number of plants that would be designed, built, and maintained below the new quality acceptance level and (2) how far below the new level the QA programs of these plants would actually operate.

Cost Estimate

Industry Cost: It was estimated that (1) the plant user cost applied to 40 plants in the design phase or under construction, (2) an average of 0.5 man-year/plant was required to develop an expanded QA list, (3) an additional 0.25 man-year/plant over 4 years was required to ensure compliance with the added QA requirements, and (4) an additional 0.1 man-year/plant would be expended to ensure compliance with the expanded QA list during the 40-year operating life of each affected plant. These estimates totaled 220 man-years and, at a rate of $100,000/man-year, the total industry cost was estimated to be $22 million (M).

NRC Cost: The NRC cost was estimated in the TMI Action Plan[48] to be 2.5 man-years or $0.25M.

Total Cost: The total industry and NRC cost associated with the solution was $(22 + 0.25)M or $22.25M.

Conclusion

Although a value/impact score was not calculated, the staff believed that the assurance of safer operation justified a high priority ranking for the issue.

The original intent of this issue was to identify those systems, structures, and components beyond those labeled "safety-related," prioritize their importance to safety, and prepare a generic QA list. This was reflected in 10 CFR 50.34(f)(3)(ii), which states, "Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR part 50 includes all structures, systems, and components important to safety. (I.F.1)." However, the staff's "Interim Reliability Evaluation Program [IREP] Procedures Guide," issued March 1983,[812] failed to identify either the need for a QA list for ITS structures, systems, and components or the basis for a generic list even if one should be needed. The first four IREP studies performed at nuclear plants were reported in NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One—Unit Once Nuclear Power Plant," issued June 1982;[366] NUREG/CR-2802, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1 Nuclear Plant," issued August 1982;[367] NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," issued April and July 1983;[810] and NUREG/CR-3511, "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," issued May and October 1984.[811] The staff's resolution of the IREP issue is discussed in Item II.C.1.

In January 1984, the NRC issued Generic Letter 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety Related,'"[1177] to clarify agency use of the terms "important to safety" and "safety related." This letter summarized the NRC's intention to pursue QA requirements for ITS equipment on a case-by-case basis; further clarification was provided in the Commission's Memorandum and Order CLI-84-9[1178] in June 1984. The first proposed rule on ITS was presented in SECY-85-119, "Issuance of Proposed Rule on the Important-to-Safety Issue," dated April 5, 1985,[1179] and was later disapproved by the Commission, which concluded that a specific listing of ITS equipment was not required to be maintained.[1180] Thus, the issue of expansion of the QA list to cover ITS equipment was considered to be closed and was not addressed in the second staff submittal on the ITS rule in SECY-86-164, "Proposed Rule on the Important-to-Safety Issue," dated May 29, 1986.[1181] Therefore, this issue was RESOLVED with no new requirements.[1182]

ITEM I.F.2: DEVELOP MORE DETAILED QA CRITERIA

Description

Historical Background

The overall objective of this TMI Action Plan[48] item was the improvement of the QA program for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities were conducted in a manner commensurate with their importance to safety. Several systems important to the safety of TMI-2 were not designed, fabricated, and maintained at a level equivalent to their safety importance. This condition existed at other plants and resulted primarily from the lack of clarity in NRC guidance. This situation and other problems relating to the QA organization, authority, reporting, and inspection were identified by the various TMI accident investigations and inquiries.[48]

Safety Significance

The intent of this item was to provide more explicit and detailed criteria concerning the elements that, in general, were found in well-conducted QA programs. Providing these more detailed criteria was expected to result in the establishment of QA programs of the caliber desired. The NRC believed that such programs would result in the detection of deficiencies in design, construction, and operation.

Possible Solutions

The proposed more detailed QA criteria for design, construction, and operations included the following:[48]

(1) Assure the independence of the organization performing the checking functions from the organization responsible for performing the tasks. For the construction phase, consider options for increasing the independence of the QA function. Include an option to require that licensees perform the entire quality assurance/quality control (QA/QC) function at construction sites. Consider using the third-party concept for accomplishing the NRC review and audit and making the QA/QC personnel agents of the NRC. Consider using the Institute of Nuclear Power Operations to enhance QA/QC independence.

(2) Include the QA personnel in the review and approval of plant operational maintenance and surveillance procedures and quality-related procedures associated with design, construction, and installation.

(3) Include the QA personnel in all activities involved in design, construction, installation, preoperational and startup testing, and operation.

(4) Establish criteria for determining QA requirements for specific classes of equipment such as instrumentation, mechanical equipment, and electrical equipment.

(5) Establish qualification requirements for QA and QC personnel.

(6) Increase the size of the licensees' QA staff.

(7) Clarify that the QA program is a condition of the construction permit and operating license and that substantive changes to an approved program must be submitted to the NRC for review.

(8) Compare NRC QA requirements with those of other agencies (i.e., National Aeronautic and Space Administration, Federal Aviation Administration, U.S. Department of Defense) to improve NRC requirements.

(9) Clarify organizational reporting levels for the QA organization.

(10) Clarify requirements for maintenance of "as-built" documentation.

(11) Define the role of QA in design and analysis activities. Obtain views on prevention of design errors from licensees, architect-engineers, and vendors.

Priority Determination

The NRC staff assumed that the above criteria would be adopted by the nuclear industry. The staff made a priority determination of the benefit of the above 11 items for improving QA. (The staff did not make a priority determination of the benefit of a QA program itself.)

To address this issue adequately, improvement in the QA program must be developed independent of the performing organization. Furthermore, the QA organization must have the confidence and the ear of higher management so that QA concerns can be heard and acted upon. The deficiency of the effort called for in this issue was that the effectiveness of the improvement program depended on the acceptance, attitudes, and emphasis given by plant management to the benefits to be derived from a QA program. Licensees that placed a high importance on QA efforts would probably be able to incorporate the intent of the QA enhancement program without making major changes to their organizational structure or in the way they performed their plant operations. However, for those licensees that continued to do business "as usual," the changes could be more cosmetic than real. They would probably seek ways to establish a QA organization that, on the surface, might appear reasonable but that, in reality, could be a "paper tiger." Enclosure 1 of SECY-82-352, "Assurance of Quality," dated August 20, 1982,[308] states the following: "In sum, the fundamental issues can best be characterized as a lack of total management commitment to quality and the uncertainty in industry's and NRC's ability to detect and correct the resulting deficiencies."

Conclusion

Although the QA improvement program could result in the establishment of an improved QA organizational structure at many plants, the results depended heavily on management acceptance. Lack of program implementation and management acceptance, rather than inadequate criteria as suggested by this issue, were the primary causes of deficiencies in QA. Increasing the detail of the QA criteria had little potential for improving the quality of design, construction, or operation and, therefore, reducing risk. Items I.F.2(2), I.F.2(3), I.F.2(6), and I.F.2(9), which addressed the concern stated above, were included in the July 1981 revision to Chapter 17 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (the SRP).[11]

The NRC believed that the issue of QA in nuclear power plants should be a high priority. However, the issue and solutions to QA deficiency as described herein (except for completed issues I.F.2(2), I.F.2(3), I.F.2(6), and I.F.2(9)) failed to address the problem of management acceptance of QA programs. Therefore, the residual items were given a LOW priority.

The NRC staff conducted a review[1964] of the seven LOW priority issues in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluations. The staff determined that the operating experience has not indicated a change in the safety significance of these issues. In addition, the staff verified that the current NRC regulatory requirements or guidance address these issues and identified the applicable regulatory framework as presented below. Because these items have been addressed by the existing regulations and the operating experience has not raised the significance of these issues, the NRC staff DROPPED these issues from further pursuit. The following section provides a discussion to demonstrate the application of the NRC regulatory framework for QA to each issue and to support their disposition.

ITEM I.F.2(1): ASSURE THE INDEPENDENCE OF THE ORGANIZATION PERFORMING THE CHECKING FUNCTION

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933, published in November 1983. In 1998, consideration of new information[1715] on the lack of independence in the checking function submitted by Region IV in April 1997 did not change this conclusion.[1716]

The staff conducted a review[1964] of this issue in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluations. According to 10 CFR 50.34(f)(3)(iii), "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions." In addition, Section 17.5 of the SRP[11] states that "the QA program requires independence between the organization performing checking functions from the organization responsible for performing the functions. (This provision applies to DC applicant, ESP, and construction QA programs. This provision is not applicable to design reviews/verifications.)"

The NRC staff concluded that this item has been adequately addressed by the NRC's regulations and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(2): INCLUDE QA PERSONNEL IN REVIEW AND APPROVAL OF PLANT PROCEDURES

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(3): INCLUDE QA PERSONNEL IN ALL DESIGN, CONSTRUCTION, INSTALLATION, TESTING, AND OPERATION ACTIVITIES

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(4): ESTABLISH CRITERIA FOR DETERMINING QA REQUIREMENTS FOR SPECIFIC CLASSES OF EQUIPMENT

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion II, "Quality Assurance Program," of Appendix B to 10 CFR Part 50 states that "The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety." In addition, applicants or license holders commit to the following standards, which identify requirements for specific classes of equipment:

- Subpart 2.4, "Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities," American Society of Mechanical Engineers (ASME) NQA-1-1994

- Subpart 2.5, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations for Nuclear Power Plants," ASME NQA-1-1994

- Subpart 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications," ASME NQA-1-1994

- Subpart 2.8, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for Nuclear Power Plants," ASME NQA-1-1994

Based on the review of NRC regulations related to this issue, the staff concluded that Item I.F.2(4) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(4) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(5): ESTABLISH QUALIFICATION REQUIREMENTS FOR QA AND QC PERSONNEL

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion II of Appendix B to10 CFR Part 50 establishes requirements for the training of personnel: "The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained." In addition, Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants,"[226] Revision 3, provides guidance that is acceptable to the NRC staff on qualifications and training for nuclear power plant personnel. This regulatory guide endorses American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.1-1993, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants,"[253] with certain clarifications, additions, and exceptions.

Moreover, 10 CFR 50.34(f)(3)(iii) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR needs to "establish a quality assurance (QA) program based on consideration of…(E) establishing qualification requirements for QA and QC personnel." Finally, Section 17.5 of the SRP[11] describes the SRP acceptance criteria for "Training and Qualification Criteria—Quality Assurance."

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(5) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(5) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(6): INCREASE THE SIZE OF LICENSEES' QA STAFF

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(7): CLARIFY THAT THE QA PROGRAM IS A CONDITION OF THE CONSTRUCTION PERMIT AND OPERATING LICENSE

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

The regulation at 10 CFR 50.54(a)(1) clearly states that implementation of the QA program is a condition in every nuclear power reactor operating license issued under 10 CFR Part 50: "Each nuclear power plant or fuel reprocessing plant licensee subject to the quality assurance criteria in appendix B of this part shall implement, under § 50.34(b)(6)(ii) or § 52.79 of this chapter, the quality assurance program described or referenced in the safety analysis report, including changes to that report. However, a holder of a combined license under part 52 of this chapter shall implement the quality assurance program described or referenced in the safety analysis report applicable to operation 30 days prior to the scheduled date for the initial loading of fuel." In addition, 10 CFR 50.54(a)(1) is also a condition in every combined license issued under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Finally, 10 CFR 52.17(a)(1)(xi), 10 CFR 52.47(a)(19), and 10 CFR 52.79(a)(25) outline the QA program requirements for applicants for early site permits (ESPs), standard design certifications (DCs) and combined licenses, respectively. SRP[11] Section 17.5 outlines a standardized QA program for DC, ESP, construction permit, operating license, and combined license applicants and holders.

Moreover, this issue specifies that "substantive changes to an approved program must be submitted to NRC for review." This part of the issue is also addressed by 10 CFR 50.54(a)(4), which states that "Changes to the quality assurance program description that do reduce the commitments must be submitted to the NRC and receive NRC approval prior to implementation." The regulation at 10 CFR 50.54(a)(4)(i)–(iv) outlines the process to make these changes.

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(7) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(7) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(8): COMPARE NRC QA REQUIREMENTS WITH THOSE OF OTHER AGENCIES

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

On July 9, 2003, the results of the staff's effort to review international quality assurance standards against the existing Appendix B to 10 CFR Part 50 framework were reported by issuance of SECY-03-0117, Approaches for Adopting More Widely Accepted International Quality Standards."[1965] In addition, approaches for adopting international quality standards for safety-related components in nuclear power plants into the existing regulatory framework were assessed. SECY-03-0117[1965] also reviewed existing NRC quality assurance requirements and efforts to improve their effectiveness and efficiency. The staff concluded in SECY-03-0117[1965] that considerable actions had already been taken or were in progress to reduce unnecessary regulatory burden on licensees resulting from compliance with Appendix B to 10 CFR Part 50 requirements. In addition, the proposed 10 CFR 50.69 risk-informed rulemaking would provide a more efficient and effective regulatory process while continuing to maintain safety. The staff evaluation of the differences between Appendix B to 10 CFR Part 50 and ISO 9001 is summarized in the attachment to SECY-03-0117.[1965]

The staff concluded that the analysis presented in SECY-03-0117[1965] addressed Item I.F.2(8) adequately and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(9): CLARIFY ORGANIZATIONAL REPORTING LEVELS FOR THE QA ORGANIZATION

This item was evaluated in Item I.F.2 above and was determined to be RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the SRP.[11]

ITEM I.F.2(10): CLARIFY REQUIREMENTS FOR MAINTENANCE OF "AS-BUILT" DOCUMENTATION

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion VI, "Document Control," and Criterion XVII, "Quality Assurance Records," of Appendix B to 10 CFR Part 50 establish requirements for issuing, identifying, and retrieving QA records. In addition, NRC-accepted practices for the collection, storage, and maintenance of QA records for nuclear power plants, independent storage of spent nuclear fuel and high-level radioactive waste facilities, special nuclear materials, packaging and transportation of radioactive materials, and gaseous diffusion plants are described in ANSI/ASME NQA-1.[1966]

Criterion VI of Appendix B to 10 CFR Part 50 describes the requirements to control changes in documents: "Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization."

Moreover, 10 CFR 50.34(f)(3)(iii) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "establish a quality assurance (QA) program based on consideration of…(G) establishing procedures for maintenance of 'as-built' documentation." Finally, Section 17.5 of the SRP[11] states that "A program is required to be established to control the development, review, approval, issue, use, and revision of documents." This section includes as-built drawings as one of the examples of controlled documents: "Examples of controlled documents include design drawings, as-built drawings, engineering calculations."

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(10) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(10) and DROPPED this item from further pursuit.[1964]

ITEM I.F.2(11): DEFINE THE ROLE OF QA IN DESIGN AND ANALYSIS ACTIVITIES

This item was evaluated in Item I.F.2 above and was determined to be a LOW priority issue in the main report of NUREG-0933, published in November 1983.

Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 describes the requirements of the program for the design of items. As explained in this criterion, measures should be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. In addition, these measures should include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. The design control measures provide for verifying or checking the adequacy of design and are applied to items such as the reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Moreover, 10 CFR 50.34(f)(3)(iii) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "establish a quality assurance (QA) program based on consideration of…(H) providing a QA role in design and analysis activities." Finally, Section 17.5 of the SRP[11] states that "The QA role in design and analysis activities is defined. Design documents are reviewed by individuals knowledgeable and qualified in QA to ensure the documents contain the necessary QA requirements. (This applies to DC applicants, ESP, and construction QA programs.)"

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that Item I.F.2(11) has been adequately addressed by the existing regulations. Therefore, the staff changed the status of Item I.F.2(11) and DROPPED this item from further pursuit.[1964]

References

11. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," U.S. Nuclear Regulatory Commission, (1st Ed.) November 1975, (2nd Ed.) March 1980, (3rd Ed.) July 1981.

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

226. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1971, (Rev. 1) September 1975 [8801130111], (Rev. 1-R) May 1977 [7907100073], (Rev. 2) April 1987 [8907180147], (Rev. 3) May 2000 [ML003706932]

253. ANSI/ANS-3.1-1993, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants."

308. SECY-82-352, "Assurance of Quality," U.S. Nuclear Regulatory Commission, August 20, 1982. [8209160068]

366. NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One—Unit One Nuclear Power Plant," U.S. Nuclear Regulatory Commission, June 1982.

367. NUREG/CR-2802, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1 Nuclear Plant," U.S. Nuclear Regulatory Commission, August 1982, (Appendix A) August 1982, (Appendix B) August 1982, (Appendix C) August 1982.

810. NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1983, (Vol. 2) August 1983, (Vol. 3) July 1983, (Vol. 4) July 1983.

811. NUREG/CR-3511, "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) May 1984, (Vol. 2) October 1984.

812. NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide," U.S. Nuclear Regulatory Commission, March 1983.

1177. Letter to All Holders of Operating Licenses, Applicants for Operating Licenses and Holders of Construction Permits for Power Reactors from U.S. Nuclear Regulatory Commission, "NRC Use of the Terms, 'Important to Safety' and 'Safety Related' (Generic Letter 84-01)," January 5, 1984. [8401050382]

1178. Memorandum and Order CLI-84-9, U.S. Nuclear Regulatory Commission, June 6, 1984. [8406070146]

1179. SECY-85-119, “Issuance of Proposed Rule on the Important-to-Safety Issue,” U.S. Nuclear Regulatory Commission, April 5, 1985. [8505030656]

1180. Memorandum for W. Dircks from S. Chilk, “Staff Requirements—SECY-85-119—‘Issuance of Proposed Rule on the Important-to-Safety Issue,’” December 31, 1985. [8601160559]

1181. SECY-86-164, “Proposed Rule on the Important-to-Safety Issue,” U.S. Nuclear Regulatory Commission, May 29, 1986. [8607010004]

1182. Memorandum for V. Stello from E. Beckjord, “Resolution of Generic Issue I.F.1, ‘Expand QA List,’” January 12, 1989. [9704150147]

1715. Memorandum for D. Morrison from T. Gwynn, “Periodic Review of Low-Priority Generic Safety Issues,” April 16, 1997. [9909290132]

1716. Memorandum for T. Gwynn from T. Martin, “Periodic Review of Low-Priority Generic Safety Issues,” July 13, 1998. [9909290134]

1964. Memorandum for B.W. Sheron from B.G. Beasley, “LOW Priority Generic Issues,” March 17, 2011. [ML092520025]

1965. SECY-03-0117, “Approaches for Adopting More Widely Accepted International Quality Standards,” July 9, 2003. [ML031490421]

1966. American Society of Mechanical Engineers, “Quality Assurance Program Requirements for Nuclear Facility Applications,” ANSI/ASME Standard NQA -1, Washington, DC.

TASK II.B: CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW

The objective of this task was to enhance public safety and reduce individual and societal risk by developing and implementing a phased program to include, in safety reviews, consideration of core degradation and melting beyond the design basis.

ITEM II.B.1: REACTOR COOLANT SYSTEM VENTS

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-10 was established by DL/NRR for implementation purposes.

ITEM II.B.2: PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-11 was established by DL/NRR for implementation purposes.

ITEM II.B.3: POST-ACCIDENT SAMPLING

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-12 was established by DL/NRR for implementation purposes.

ITEM II.B.4: TRAINING FOR MITIGATING CORE DAMAGE

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-13 was established by DL/NRR for implementation purposes.

ITEM II.B.5: RESEARCH ON PHENOMENA ASSOCIATED WITH CORE DEGRADATION AND FUEL MELTING

The three parts of this item are evaluated below.

ITEM II.B.5(1): BEHAVIOR OF SEVERELY DAMAGED FUEL

Items II.B.5(1) and II.B.5(2) were combined and evaluated together under Item II.B.5(1).

Description

Historical Background

For a number of key severe accident sequences, there are critical phenomenological unknowns or uncertainties that impact containment integrity assessments and judgments regarding the desirability of certain mitigating features. The phenomena fall into three broad categories: (1) the behavior of severely damaged fuel, including oxidation and H_2 generation; (2) the behavior of the core-melt in its interaction with water, concrete, and core-retention materials; and (3) the effect of potential H_2 burning and/or explosions on containment integrity. Steam explosions were also to be considered in this category. Previous work in these several areas received less attention since they related to accidents beyond the design basis. At the time this TMI Action Plan[48] item was raised, RES was conducting major programs to support the basis for rulemaking and to confirm certain licensing decisions. Complementary efforts conducted within NRR were to address specific licensing issues related to the subject research.

(1) Behavior of Severely Damaged Fuel

(a) In-Pile Studies: Fuel behavior research was to include in-pile testing to help evaluate the effects of conditions leading to severe fuel damage. Such tests were being performed in the INEL Power Burst Facility (PBF) in FY 1983 and later in the Annular Core Research Reactor (ACRR) at SNL and in the NRU reactor at Chalk River National Lab, Canada. In the PBF, RES was to perform a series of in-reactor fuel experiments to determine the effect of heating and cooling rates on damage to the bundle, rod fragmentation, distortion, and debris formation. Fission product release and H_2 generation were also to be measured during the testing. Separate effects studies were to be conducted on rubble beds in the ACRR at SNL.

(b) Hydrogen Studies: The objective of this work was to increase the understanding of the formation of H_2 in a reactor from metal-water reactions, radiolytic decomposition of coolant, and corrosion of metals, and to determine its consequences in terms of pressure-time histories and H_2 deflagration or detonation. This work was also to include: (1) the preparation of a compendium of information related to H_2 as it affects reactor safety; (2) analysis of radiolysis under accident conditions; (3) a review of H_2 sampling and analysis methods; (4) a study of the effects of H_2 embrittlement on reactor vessel materials; and (5) a review of means of handling accident-generated H_2 with recommendations on improving existing methods. Results of these studies were considered in the resolution of Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment," and were not considered further in this issue.

(c) Studies of Post-Accident Coolant Chemistry: The RES objective in this area was the development of a relationship between fission product release and fuel failure and the improvement of post-accident sampling and analysis techniques. This was to be accomplished by the investigation of fission product release in a variety of fuel failure experiments.

(d) Modeling of Severe Fuel Damage: The effort in this area was the development of models for fuel rods operating beyond 2200^0F that suffer a loss in geometry in order to compute extensive damage phenomena (such as eutectic liquid formation, fuel slumping, oxidation, and H_2 generation, fission product release and interaction with the coolant, rubble-bed particle size, extent of fuel and clad melting, and flow blockage).

(2) Behavior of Core-Melt

The RES fuel-melt research program was to develop a base and verified methodology for assessing the consequences and mitigation of fuel-melt accidents. The program addressed the range of severe reactor accident phenomena from the time when extensive fuel damage and major core geometry changes occur until the containment has failed and/or the molten core materials have attained a semi-permanent configuration and further movement is terminated. Studies of improvements in containment design to reduce the risk of core-melt accidents were also included.

The program was composed of integrated tasks that included scoping, phenomenological and separate effects tests, and demonstration experiments that provided results for the development and verification of analytical models and codes. These codes and supporting data were then used for the analysis of thermal, mechanical, and radiological consequences of accidents and for decisions related to requirements of design features for mitigation and performance confirmation. The technical scope of the program included work in the following areas: fuel debris behavior; fuel interactions with structure and soil; radiological source term; fuelcoolant interactions; systems analysis codes; and mitigation features.

Safety Significance

The results of the research programs described above were expected to find broad application in areas such as PRA, accident analysis, siting, evacuation planning, emergency procedures, code development, etc. Thus, these programs would have considerable value just as licensing improvement efforts. However, the programs had sufficiently well-defined scopes to permit some estimates of direct safety significance. These programs were directed at a better understanding of severely damaged and molten cores. Once a core is in this state, any safety significance has to be in the area of minimizing radioactive releases and consequent dose to the public.

Possible Solutions

It was assumed that means would be devised to reduce the probability of containment failure and release of activity to the environment. Completely different approaches could be suggested after the results of the research programs were known.

The "classical" engineering approaches to handling degraded or melted cores are filtered vents to prevent containment overpressure and core-retention devices (core catchers) to prevent containment basemat melt-through. These approaches were used for cost estimates, but the other priority parameters were not specific to these approaches.

Priority Determination

Studies[64] of this issue by PNL considered only containment basemat melt-through. The approach presented here was expanded to include other aspects. The effect on a PWR with a dry containment was considered, based partly on the availability of information. It was not expected that the results for other containments or for BWRs would be greatly different, at least in the context of the uncertainty of such an analysis.

Frequency Estimate

Essentially, all core-melts are assumed to result in containment failure in WASH-1400.[16] To estimate the effect of being able to deal with a severely damaged core, this assumption was relaxed. The modes of containment failure for PWRs were defined as follows:

α - containment rupture due to a reactor vessel steam explosion

β - containment failure due to inadequate isolation of openings and penetrations

γ - containment failure due to H_2 burning

δ - containment failure due to overpressure

ε - containment vessel melt-through

Assuming that the research programs were successful in leading to engineering solutions, reductions in the frequency of the various failure modes were estimated as follows:

α - 10% (Little can be done about steam explosions.)

β - 0% (This does not affect isolation failure.)

γ,δ - 90% (Venting containment should be quite effective if methods are available for sizing the vent and determining what filtration is needed.)

ε - 90% (Should be achievable if a core catcher can be designed.)

Consequence Estimate

The consequences were straightforward in the sense that the consequences of each release category have been studied. However, the reduction in consequences was more difficult to assess since the release from a molten core in a tight containment is still not zero. Instead, it depends on the containment design leak rate, the efficiency of filtration of a containment relief vent, etc. To allow for this, it was assumed instead that the prevented releases corresponding to the α, γ, δ, and ε failure modes were similar to a PWR-9 release. The results of this calculation are summarized in Table II.B-1. For a new (forward-fit) plant (which was the most likely candidate for implementation), the public risk reduction was estimated to be 1,600 man-rem.

Cost Estimate

Industry Cost: PNL estimated[64] the cost of a core retention device to be \$1.4M for a forward-fit. SNL estimated[312] the cost of a filtered containment vent to be on the order of a few million dollars. Thus, the industry cost was projected to be \$10M/reactor.

NRC Cost: PNL estimated[64] the NRC cost to be \$2.3M, assuming implementation at 134 reactors. In reality, implementation might take place at a far smaller number of plants due to considerations of containment type, backfit vs. forward-fit, etc. However, even if only 10 plants were affected, the NRC cost would be insignificant compared to licensee costs. Therefore, NRC costs were neglected.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be $10M/reactor.

Table II.B-1

Release Category	Frequency* $(RY)^{-1}$	% Reduction**in Frequency	$\Delta F\ (RY)^{-1}$	R (man-rem)	FR
PWR-1	5.3×10^{-8}	10%	5.3×10^{-9}	4.9×10^{6}	2.6×10^{-2}
PWR-2	6.7×10^{-6}	90%	6.0×10^{-6}	4.8×10^{6}	2.9×10^{1}
PWR-3	2.6×10^{-6}	81%	2.1×10^{-6}	5.4×10^{6}	1.1×10^{1}
PWR-4	2.1×10^{-11}	-	-	2.7×10^{6}	-
PWR-5	4.9×10^{-8}	-	-	1.0×10^{6}	-
PWR-6	6.3×10^{-7}	90%	5.7×10^{-7}	1.4×10^{5}	8.0×10^{-2}
PWR-7	3.4×10^{-5}	90%	3.1×10^{-5}	2.3×10^{3}	7.1×10^{-2}
PWR-8	8.0×10^{-7}	-	-	7.5×10^{4}	-
PWR-9	4.0×10^{-4}	-	-3.9×10^{-5}	1.2×10^{2}	-4.7×10^{-3}
			TOTAL:		4.0×10^{1}

* Because the specific containment failure mode was of interest here, the frequencies above were "unsmoothed." This is in contrast to the calculations in WASH-1400[16] which assumed a 10% contribution in frequency from adjacent release categories.

** Release Category PWR-1 is made up entirely of α failures and thus was assigned a 10% reduction in frequency. Categories PWR-2, PWR-6, and PWR-7 are made up of γ, δ, and ε failures and were thus assigned 90%. Category PWR-3 contains both α and δ failures which results in a net assignment of 81%.

Value/Impact Assessment

Based on a potential public risk reduction of 1,600 man-rem/reactor and a cost of $10M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{1{,}600 \text{ man-rem/reactor}}{\$10\text{M/reactor}}$$

$$= 160 \text{ man-rem/\$M}$$

Conclusion

Based on the factors considered above, this issue was given a high priority ranking (see Appendix C). However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing

Issue.[1102] The issue was pursued[138l] as part of SARP Issue L2, "In-Vessel Core Melt Progression and Hydrogen Generation," documented in NUREG-1365.[1382]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issue. Because licensing and regulatory impact issue are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM II.B.5(2): BEHAVIOR OF CORE-MELT

This item was evaluated in Item II.B.5(1) above and determined to be a high priority (see Appendix C). However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[138l] as part of SARP Issue L2, "In-Vessel Core Melt Progression and Hydrogen Generation," documented in NUREG-1365.[1382]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM II.B.5(3): EFFECT OF HYDROGEN BURNING AND EXPLOSIONS ON CONTAINMENT STRUCTURE

Description

Historical Background

TMI Action Plan[48] Item II.B.5 called for research into the phenomena associated with severe core damage and core melting. Item II.B.5(3) addressed the effect of H_2 burns and/or explosions on containment integrity.

Safety Significance

Whereas Items II.B.5(1) and II.B.5(2) dealt with (among other things) the generation of H_2 via radiolysis, metal-water interaction, interaction of a molten core with concrete, etc., Item II.B.5(3) was concerned with the effects on the containment of the burning and/or detonation of this H_2. If the containment retains its integrity, even a severe accident resulting in a damaged or molten core produces relatively low offsite consequences. Item II.B.5(3) also included the effect of steam explosions. Again, the emphasis here was not in preventing the explosion but, instead, in maintaining containment integrity.

Possible Solution

Most of the work on Item II.B.5(3) was couched in terms of a stronger containment.

Priority Determination

Item II B.5(3) was, to a large extent, similar to Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment." Issue A-48 was somewhat more general in that it included the effects of a H_2 burn or detonation on containment penetrations and on safety systems located within the containment, not just the structural response of the containment. In addition, Issue A-48 included measures for control of the H_2 burn and thus had preventive as well as mitigative aspects. However, even though Issue A-48 was expected to use the results of Item II.B.5(3), Item II.B.5(3) was not integrated into Issue A-48 because: (1) the scope of Issue A-48 was still under discussion; and (2) Item II.B.5(3) included steam explosions as well as H_2 burns.

Frequency/Consequence Estimate

In WASH-1400,[16] the PWR sequences refer to steam explosion-induced containment failures as "α" failures; containment failures induced by an H_2 burn are called "γ" failures. Sequences including these two failure modes can be found in Release Categories PWR-1, PWR-2, and PWR-3. It was assumed that the possible solution would result in a 90% reduction in the probabilities of the sequences involving these two failure modes. The results are tabulated in Table II.B-2 below.

TABLE II.B-2

Release Category (F)	$^{\alpha}$Frequency (per RY)	$^{\gamma}$Frequency(F) (per RY)	Consequences(R)(man-rem)	0.9FR (man-rem/RY)
PWR-1	5.3×10^{-8}	-	4.9×10^{6}	0.23
PWR-2	-	7.0×10^{-7}	4.8×10^{6}	3.00
PWR-3	3.4×10^{-7}	-	5.4×10^{6}	1.70
PWR-7	-3.9×10^{-7}	-7.0×10^{-7}	2.3×10^{3}	- 0.002
			TOTAL:	4.9

The PWR-7 category has a negative contribution because a molten core still gives some release, even if containment failure is prevented. Thus, it was assumed that the events which would have been α or γ failures instead lead to PWR-7 releases.

Over a 40-year plant life, the risk reduction above corresponded to about 200 man-rem/reactor. This was calculated using WASH-1400[16] data for a PWR with a large, dry containment. BWR pressure-suppression containments and PWR ice-condenser containments have a much smaller free volume and thus are more susceptible to α and γ failures. Therefore, the risk for these plants could well be considerably higher.

Cost Estimate

Industry Cost: Without the results of research at the time of this evaluation, it was difficult to assess costs. A stronger containment could cost $15M, based on doubling the 3.5 foot wall thickness of a (150 ft x 200 ft) structure. (Such structures cost roughly $1,000/cubic yard of concrete.)

NRC Cost: NRC costs were considered to be negligible.

Total Cost: The total industry and NRC cost associated with the possible solution was $15M/reactor.

Value/Impact Assessment

Based on an estimated public risk reduction of 200 man-rem/reactor and a cost of $15M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{200 \text{ man-rem/reactor}}{\$15\text{M/reactor}}$$

$$= 13 \text{ man-rem/\$M}$$

Conclusion

The public risk estimate for this issue was significant even for dry containments. Because of the difficulty in determining a cost-effective solution, the issue was given a medium priority ranking (see Appendix C). However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[1381] as part of SARP Issue L3, "Hydrogen Transport and Combustion," documented in NUREG-1365.[1382]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM II.B.6: RISK REDUCTION FOR OPERATING REACTORS AT SITES WITH HIGH POPULATION DENSITIES

Description

Historical Background

This TMI Action Plan[48] item involved the review of operating reactors in areas of high population density to determine what additional measures and/or design changes could be implemented that would further reduce the probability of a severe reactor accident, and would reduce the consequences of such an accident by reducing the amount of radioactive releases and/or by delaying any radioactive releases, and thereby provide additional time for evacuation near the sites.

Risk studies were completed in 1981 for the Zion and Limerick sites and in 1982 for Indian Point. Although risk assessments of other sites were conducted by other NRC programs, e.g., National Reliability Evaluation Program (NREP), no further risk studies were envisioned as part of this issue. Further efforts directed towards this issue were review of the analyses and the possible implementation of site-specific fixes to reduce the risk at these sites. Special hearings were scheduled in FY 1982 to review possible design changes for Indian Point and follow-up work in connection with the accepted fixes was anticipated following these hearings.

Safety Significance

Concern existed over the potential for above-average societal risk due to accidents at reactor sites located near regions of high population densities.

Possible Solutions

As mentioned above, hearings were scheduled on possible fixes at the Indian Point site to reduce risk. The actual fixes that resulted from these hearings were unknown at the time of this evaluation. Nevertheless, it was assumed that fixes would be made to reduce the likelihood of the most dominant accident sequences contributing to the frequency of core-melt accidents.

Priority Determination

Assumptions

Based on a review of similar Reactor Safety Study Methodology Application Program (RSSMAP) and Interim Reliability Evaluation Program (IREP) analyses, it was assumed that two sequences contributed to a large portion (50%) of the likelihood of a core-melt. It was further assumed that it was possible to reduce the frequency of each sequence by a factor of 10.

Frequency Estimate

Based on age and other related factors, it was believed that reactors in this category had an increased frequency of core-melt over the baseline plant (Oconee-3) by a factor of 5.5. Thus, the revised baseline core-melt frequency (F) was given by:

$$\begin{aligned} F &= (5.5)(8.2 \times 10^{-5}/\text{RY}) \\ &= 4.5 \times 10^{-4}/\text{RY} \end{aligned}$$

Assuming that the dominant sequences (50% of the frequency) could be reduced by a factor of 10, the revised core-melt frequency was $(0.55)(4.5 \times 10^{-4})/\text{RY} = 2.5 \times 10^{-4}/\text{RY}$.

Consequence Estimate

Considering the same factors used above to estimate the core-melt frequency, the affected plants would have an exposure increase over the mean population density (340 persons/square-mile) and release fractions by a factor of 3. Thus, this exposure increase (R) was given by:

$$R = (3)(2.5 \times 10^6 \text{ man-rem})$$
$$= (7.5 \times 10^6) \text{ man-rem}$$

The baseline public risk was $(4.5 \times 10^{-4}/RY)(7.5 \times 10^6$ man-rem) or 3,380 manrem/RY. The revised public risk was $(2.5 \times 10^{-4}/RY)(7.5 \times 10^6$ man-rem) or 1,880 man-rem/RY. The resulting change in public risk was then 1,500 man-rem/RY resulting from the reduction in core-melt frequency of $2 \times 10^{-4}/RY$. Over the estimated 27 years of remaining plant life, this would result in a total risk reduction of 40,500 man-rem/reactor.

Cost Estimate

Industry Cost: Licensee costs were estimated to be $4M/reactor to implement the changes required to reduce the two dominant sequences.

NRC Cost: NRC costs were estimated to be $22,000.

Total Cost: The total industry and NRC cost associated with the possible solution was $(4 + 0.02)M/reactor or $4.02M/reactor.

Value/Impact Assessment

Based on an estimated public risk reduction of 40,500 man-rem/reactor and a cost of $4.02M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{40{,}500 \text{ man-rem/reactor}}{\$4.02\text{M/reactor}}$$

$$= 10{,}000 \text{ man-rem/\$M}$$

Other Considerations

The accident avoidance cost was estimated to be approximately $11M which would result in a potential cost saving of $7M, considering the $4M implementation costs.

Conclusion

Based on the above value/impact score, this issue was given a high priority ranking (see Appendix C). A staff review of PRAs submitted by the affected licensees was used to identify the strengths and weaknesses of the various plants and to assess the risk associated with their operation. A special adjudicatory proceeding was held from 1982 to 1983 during which time the issues regarding continued operation and risk of the Indian Point plants were heard. Following these hearings, the Commission concluded that neither shutdown of Indian Point Units 2 or 3 nor imposition of additional remedial actions beyond those already implemented by the licensees were warranted.[806]

The staff also reviewed the Zion PRA and concluded that the risk posed by the Zion plants was small. The dominant contributors to severe accidents at the Zion plants were examined and the staff recommended that: (1) the integrity of the two motor-operated gate valves in the RHR suction line from the RCS be checked each refueling outage; and (2) the diesel-driven containment spray pump be modified so that it could be capable of operating without AC power.[806] Thus, this item was RESOLVED and new requirements were established. DL/NRR was responsible for managing the implementation of the above recommendations.[806]

ITEM II.B.7: ANALYSIS OF HYDROGEN CONTROL

Description

The TMI-2 accident resulted in a metal-water reaction which involved H_2 generation in excess of the amounts specified in 10 CFR 50.44. As a result, it became apparent to the NRC that additional H_2 control and mitigation measures would have to be considered for all nuclear power plants. The purpose of this TMI Action Plan[48] item was to establish the technical basis for the interim H_2 control measures on small containment structures and to establish the basis for continued operation and licensing of plants, pending long-term resolution of the H_2 control issue.

Conclusion

The long-term resolution of this issue was accomplished by rulemaking as part of Item II.B.8. A final rule was published on December 2, 1981 requiring inerting of the small BWR MARK I and II containments. In addition, based on Commission guidance, interim H_2 control systems were required as a licensing condition for the intermediate volume Ice Condenser and MARK III containments. A proposed rule was published on December 23, 1981 (Federal Register 46 FR 62281) which required these systems for the intermediate volume containments. Except for pending construction permit (CP) and manufacturing license (ML) applications, no additional requirements for H_2 control or H_2 analyses were imposed at that time for large, dry containments. However, the proposed rule required that dry containments be analyzed to determine their ability to accommodate the release of large quantities of H_2 (75% metal-water reaction). Also, H_2 control requirements were established as part of the final Near-Term CP and ML Rule published on January 15, 1982.

Based on the accomplishments above, the basis for continued operation and licensing of plants with respect to the H_2 control issue was established. Future work related to finalizing the proposed rule dealing with intermediate volume containments (Ice Condenser and MARK III) and large, dry containments continued as part of Item II.B.8.

ITEM II.B.8: RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

Description

Historical Background

In the past, safety reviews concentrated on how to prevent a core from being damaged. Consequently, little attention was given to how a severely damaged core could be dealt with after damage occurred. Other subtasks within Task II.B were concerned with the study of the characteristics of degraded and melted cores (research programs) plus some immediate actions to be taken at plants in operation. Item II.B.8 envisioned both a short-term and a long-term rulemaking to establish policy, goals, and requirements to address accidents resulting in core damage greater than the existing design basis.

Item II.B.8 included an Advance Notice of Proposed Rulemaking (ANPRM) and an Interim Rule. The ANPRM was issued on December 2, 1980 (45 FR 65474) and the Interim Rule was issued in two parts: the first was issued in effective form in October 1981 (46 FR 58484) and the second was issued as a proposed rule on December 23, 1981 (46 FR 62281).

On January 4, 1982, SECY-82-1[309] was forwarded to the Commission requesting reconsideration of the approach to long-term rulemaking. The events which prompted this request were as follows:

- The Commission had required more protection from severe accidents in some licensing actions (e.g., Sequoyah) than was envisioned in the TMI Action Plan.[48]

- A rule was developed to specify additional requirements for pending CP and ML applications. Again, these requirements were somewhat more extensive than that envisioned in the TMI Action Plan.[48]

- New probabilistic risk assessments (PRAs) indicated lower risk than was previously estimated for large, dry PWR containments.

- The safety of existing plants had been considerably improved by the modifications mandated by NUREG-0737.[98]

- The industry initiated a program to study the costs and benefits of design features for mitigating severe accidents.

- An extensive research program to study damaged and melted core behavior was underway.

- A safety goal statement, based on PRA, was developed.

The substance of SECY-82-1[309] was that the uncertainty associated with long-term rulemaking was an inhibiting force on the industry. The paper then recommended that, since new applications were to be standardized, licensing could proceed on these standardized designs using the information available. PRAs and the safety goal would be used to assess plant safety. If plants needed safety features beyond the existing requirements to meet the safety goal, they could be included. This approach would not need rulemaking specifically directed at severe accident mitigation.

The Commission directed[310] the staff to make several changes recommended in SECY-82-1.[309] The staff then submitted revised papers SECY-82-1A[311] and SECY-82-1B[1405] that incorporated the changes directed by the Commission, including ACRS input. The evaluation of this item included consideration of Item II.B.7.

Safety Significance

Most of the engineered safety features at nuclear power plants of the existing generation were intended to prevent severe core damage. Relatively little attention was given in the past to dealing with a severely damaged or melted core. Once a core is damaged, the containment will still prevent the release of large amounts of radioactive material. However, once the core melts, the containment is likely to fail (although the hazard to the public varies widely, depending on the way in which the containment fails).

The degraded-core accident rulemaking was intended to require means for dealing with a damaged core. This translated into preventing the release of radioactivity and providing means for recovering from the accident. Specific items to be considered included the following: use of filtered, vented containment; H_2 control measures; core retention devices ("core catchers"); re-examination of design criteria for decay heat removal and other systems; post-accident recovery plans; criteria for locating highly radioactive systems; effects or accidents at multi-unit sites; and comprehensive review and evaluation of related guides and regulations.

Priority Determination

The safety significance of this issue was essentially the same as that of the research programs described in the analyses of Items II.B.5(1) and II.B.5(2) above. Examination of the estimated frequency of core damage and/or core-melt, coupled with estimates of the potential effectiveness of engineering solutions (and their cost) led to the recommended high priority for Items II.B.5(1) and II.B.5(2). In the same manner, Item II.B.8 had the potential for a significant (and cost-effective) reduction in public risk. In addition, it should be noted that some of the plant modifications contemplated were far more expensive to backfit than to forward-fit. Unnecessary delay could have reduced the costeffectiveness of the resolution to this issue.

Conclusion

Based on the above evaluation, this item was given a high priority ranking (see Appendix C). Work performed by RES on the H_2 control aspect of the issue resulted in a Hydrogen Control Rule that was approved by the Commission and published in the Federal Register on January 25, 1985.[807] The severe accident portion of the issue was addressed in April 1983 by a Policy Statement that set forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents (48 FR 16014). Certain severe accident technical issues identified under the discussion of long-term rulemaking were to be dealt with for future and existing plants through procedures and ongoing severe accident programs identified in the Policy Statement and described more fully in Chapter IV of NUREG-1070.[809] Thus, with the issuance of the rule on H_2 control, this item was RESOLVED and new requirements were established.[808]

References

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

16. WASH-1400 (NUREG-75/014), “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,” U.S. Atomic Energy Commission, October 1975.

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

309. SECY-82-1, "Severe Accident Rulemaking and Related Matters," U.S. Nuclear Regulatory Commission, January 4, 1982. [8201190416]

310. Memorandum for W. Dircks from S. Chilk, "Staff Requirements—Briefing on Status and Plan for Severe Accident Rulemaking (SECY-82-1)," January 29, 1982. [8202160202]

311. SECY-82-1A, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," U.S. Nuclear Regulatory Commission, July 16, 1982. [8208040432]

312. NUREG/CR-0165, "A Value-Impact Assessment of Alternate Containment Concepts," U.S. Nuclear Regulatory Commission, June 1978.

806. Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan, Task II.B.6, 'Risk Reduction for Operating Reactors at Sites With High Population Densities,'" September 25, 1985. [8510030342]

807. Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan Task II.B.8 'Rulemaking Proceeding on Degraded Core Accidents—Hydrogen Control,'" July 19, 1985. [8508010066]

808. Memorandum for W. Dircks from H. Denton, "Close Out of TMI Action Plan, Task II.B.8," August 12, 1985. [8508210316]

809. NUREG-1070, "NRC Policy on Future Reactor Designs," U.S. Nuclear Regulatory Commission, July 1985.

1102. Memorandum for H. Clayton from B. Sheron, "Criteria for Initiating Feed and Bleed," September 13, 1985. [8509180314]

1381. Memorandum for W. Minners from B. Sheron, "Update of Generic Issue Management Control System (GIMCS)," July 5, 1991. [9312220300]

1382. NUREG-1365, "Revised Severe Accident Research Program Plan," U.S. Nuclear Regulatory Commission, August 1989, (Rev. 1) December 1992.

1405. SECY-82-1B, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," U.S. Nuclear Regulatory Commission, November 24, 1982. [8301120513]

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

TASK III.D.2: PUBLIC RADIATION PROTECTION IMPROVEMENT

The objective of this task was to improve public radiation protection in the event of a nuclear power plant accident by improving (1) radioactive effluent monitoring, (2) the dose analysis for accidental releases of radioiodine, tritium, and carbon-14, (3) the control of radioactivity released into the liquid pathway, (4) the measurement of offsite radiation doses, and (5) the ability to rapidly determine offsite doses from radioactivity release by meteorological and hydrological measurements so that population-protection decisions can be made appropriately.

ITEM III.D.2.1: RADIOLOGICAL MONITORING OF EFFLUENTS

The three parts of this item were combined and evaluated together.

Description

Historical Background

This Three Mile Island (TMI) Action Plan[48] item required development and implementation of acceptance criteria for monitors used to evaluate effluent releases under accident and postaccident conditions. Criteria were to be developed for pathways to be monitored (stack, plant vent, steam dump vents) as well as for monitoring instrumentation. This was seen to encompass the requirements in Recommendation 2.1.8-b of NUREG-0578, “TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations,” issued July 1979,[57] and Appendix 2 to NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.”[224]

Liquid effluents were not envisioned as posing a major release pathway because licensees typically had installed, or were installing, adequate storage capacity to prevent discharges. Consequently, existing liquid effluent monitoring systems were considered to be adequate.

Safety Significance

This issue had no impact on core-melt accident frequency.

Possible Solution

The envisioned monitoring system would provide automatic online analysis of airborne effluents, including isotopic analyses of particulate, radioiodine, and gas samples. To prevent saturation of detectors, an automatic sample cartridge changeout feature would be included. The system would include microprocessor control and real-time readouts and would be located in a low postaccident background area. The sampling system would be designed to provide a representative sample under anticipated accident release conditions.

A pressurized-water reactor (PWR) steam-dump sampling and monitoring system would be provided for PWR safety relief and vent valves. Such a system might consist of a noble gas monitor and a radioiodine sampling and monitoring system. The features of such a system would be similar to the above-described airborne effluent monitor with two notable differences: (1) the system would be required to function in a very high humidity (steam-air mixture) environment, and (2) operation would only be required during actual steam venting. Because such venting is usually of a short-term or intermittent duration, the monitoring system activation could be keyed to the opening of the vents.

Priority Determination

Assumptions

It was assumed that improved radiological monitoring of airborne effluent would result in a reduction of public risk. The following section presents the U.S. Nuclear Regulatory Commission (NRC) staff analysis for prioritizing this issue, which was performed in 1998. This analysis, which includes frequency, consequence, and cost estimates and a value/impact assessment, has not been updated in the 2011 revision of this issue.

Frequency/Consequence Estimate

The magnitude of public risk reduction attributable to improved radiological monitoring of airborne effluents was not certain, but it was estimated by Pacific Northwest Laboratory (PNL)[64] to range from 0 to 1 percent, based on the following logic.

Existing radiological monitoring requirements, as contained in NUREG-0737, "Clarification of TMI Acton Plan Requirements,"[98] require real-time noble gas monitoring with sampling and laboratory analysis capabilities for radioiodines and particulates. Design-basis conditions defined in NUREG-0737[98] (100 microcuries per cubic centimeter radioiodines and particulates, 30-minute sample time) indicated that sample collection devices would pose special handling and analysis problems due to very high radioactivity buildup. Consequently, licensees typically provided alternate sample collection and analysis procedures. Execution of those procedures was estimated to require between 2 and 3 hours. During this time, radioiodine and particulate releases would be estimated based on computer-modeled interpretation of noble gas monitor readings, or on previous postaccident containment atmosphere analysis results, if such results were available. Public protective action recommendations would be made based on modeled estimates rather than actual effluent data. It was assumed that these recommendations would err on the conservative side (e.g., evacuate when not really required), due to the conservatism built into the modeled source terms for radioiodine and particulate releases.

Requiring licensees to have more sophisticated airborne effluent monitors would reduce the time required to obtain actual radioiodine and particulate release data to 15 minutes and essentially eliminate reliance on conservative theoretical release models extrapolated from noble gas monitor readings. As projected by the possible solution, real-time isotopic monitoring would save nearly 2 hours in arriving at realistic protective action recommendations based on actual releases.

Under these circumstances, the public risk reduction would be directly attributed to the decrease in public radiation exposure that would result from a more rapid assessment of the radioactive releases (about a 2-hour savings in analysis time). In addition, public risk may be reduced as a result of nonevacuation. The need for evacuation (presumed to exist if release knowledge was based only on noble gas monitor data) could be eliminated as a result of better knowledge of the isotopic releases. Nonevacuation would result in fewer evacuation-related risks (e.g., traffic accidents), the avoidance of which may outweigh the radiation exposure received. However, it was assumed that the public risk reduction would result primarily from the first effect (decrease in exposure due to more rapid assessment).

While protective actions can be recommended based on effluent releases in progress, the probability for a core-melt scenario was such that actions would be recommended based on anticipated releases, before the actual releases themselves. Under this assumption, monitoring effluent releases would have little or no impact on public risk and would be mainly for confirmation and quantification. This possible solution would not impact core-melt accident frequency.

At the time of this evaluation, there were 134 plants affected by the issue: 71 operating (47 PWRs and 24 boiling-water reactors (BWRs)) and 63 planned (43 PWRs and 20 BWRs). It was assumed that the average remaining plant life was 27.4 years for the 44 BWRs and 28.8 years for the 90 PWRs. The dose factors for PWR Release Categories 1 through 7 and BWR Release Categories 1 through 4 were assumed to be affected by the possible solution. From NUREG/CR-2800, “Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development,”[64] a 1-percent decrease in the dose factors resulted in an estimated total public risk reduction of 8,500 man-rem for all plants. Assuming a decrease in the dose factors of 0.5 percent for this issue, the estimated public risk reduction was 4,250 man-rem.

Cost Estimate

Industry Cost: The industry cost for equipment development, installation, support facilities, and construction was estimated to be $600,000/plant. Development of procedures, software, and calibration for the equipment was estimated to require 16 man-weeks of effort, with an additional 4 man-weeks for the initial training of all licensee operators and health physics personnel. This was estimated to add $45,400/plant to the implementation cost. Based on an estimated cost of $645,000/plant for labor and equipment, the industry cost for implementing the possible solution was (134 plants)($645,000/plant) or $86.5 million (M).

The recurring industry operation and maintenance costs were estimated at 2 man-weeks/plant-year for retraining, 1 man-week/plant-year for calibration, and a reduction of 1 man-week/plant-year (reduced laboratory analyses due to a fully automated system) for a net increase of 2 man-weeks/plant-year, or an increased cost of $4,540/plant-year. As a result, industry costs for labor and material associated with operation and maintenance of the possible solution were estimated to be $17.2M.

Thus, the total industry cost associated with this issue was $(86.5 + 17.2)M or $103.7M.

NRC Cost: The NRC cost was assumed to be limited to implementation costs for development and plant installation. Because it was assumed that the new radiological monitoring systems would require no periodic inspection effort beyond that required for current systems, no additional NRC operation cost was envisioned. The NRC development cost included 1.5 man-years and $200,000 for research, criteria development, and engineering development, for a total cost of $350,000. The NRC administrative and technical effort associated with the review and approval of licensee submittals was estimated at 0.3 man-week/plant for a total cost of $91,000 for all plants. Therefore, the total NRC cost associated with this issue was $441,000.

Total Cost: The total industry and NRC cost associated with the possible solution was $(103.7 + 0.441)M or $104.1M.

Value/Impact Assessment

Based on an estimated public risk reduction of 4,250 man-rem and a cost of \$104.1M for a possible solution, the value/impact score was given by the following:

$$S = \frac{4{,}250 \text{ man-rem}}{\$104.1\text{M}}$$

$$= 41 \text{ man-rem/\$M}$$

Other Considerations

It was anticipated that improvement of radiological monitoring of airborne effluents would have no significant impact on occupational risk. The dose required to install equipment would probably not exceed 0.5 man-rem, which was negligible compared to the typical 600 man-rem/year required to operate a plant. Minor man-rem savings might occur under accident conditions due to better direction of field survey teams; however, such savings would be negligible compared to the 19,900 man-rem total associated with response and cleanup following an accident.

Based on an estimated occupational dose of 0.5 man-rem/plant for implementation of the possible solution in 71 operating plants, the total risk increase was 36 man-rem for all plants. Inclusion of this factor into the above calculation would reduce the value/impact score.

There was no accident avoidance cost for the resolution of this issue because improved radiological effluent monitoring systems would have no impact on accident frequency or cleanup and refurbishing costs.

Conclusion

Based on the risk reduction potential and value/impact score, the issue was given a LOW priority ranking (see Appendix C) in November 1983. NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," issued December 1991,[1563] concluded that consideration of a 20-year license renewal period could change the ranking of the issue to medium priority. Further prioritization in 1995, using the conversion factor of \$2,000/man-rem approved[1689] by the Commission in September 1995, resulted in an impact/value ratio (*R*) of \$24,390/man-rem, which did not change the priority ranking. In 2010, the staff reviewed three parts of this issue in accordance with the SRM 871021A, "Staff Requirements—Briefing on Status of Unresolved Safety/Generic Issues," dated November 6, 1987,[1980] which directed the staff to conduct periodic reviews of existing LOW-priority issues to determine whether any new information was available that would necessitate reassessment of the original prioritization evaluations.[1964] Based on this review, the status of these issues was changed as described below.

ITEM III.D.2.1(1): EVALUATE THE FEASIBILITY AND PERFORM A VALUE/IMPACT ANALYSIS OF MODIFYING EFFLUENT-MONITORING DESIGN CRITERIA

The overall objective of this issue, which "is to provide assurance that all possible accident effluent-release pathways are monitored and that monitors will perform properly under accident conditions," is covered by General Design Criterion (GDC) 64, "Monitoring Radioactivity Releases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the

Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." GDC 64 states that "Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents." Moreover, 10 CFR 50.34(f)(2)(xvii)(E) establishes the requirement for monitoring noble gas effluents and continuous sampling of radioactive iodines and particulates in gaseous effluents. According to this part of the regulation, "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982," in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR, needs to "Provide instrumentation to measure, record and readout in the control room:…(E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples." Finally, 10 CFR 50.34(f)(2)(xxvii) and (2)(xxviii) establish requirements for monitoring of inplant radiation and airborne radioactivity for a broad range of routine and accident conditions and for evaluating potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions.

In addition to the regulations stated above, Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (the SRP),[11] states that "Provisions should be made for the installation of instrumentation and monitoring equipment and/or sampling and analyses of all normal and potential effluent pathways for release of radioactive materials to the environment, including nonradioactive systems that could become radioactive through interfaces with radioactive systems." Table 1 of Section 11.5 of the SRP[11] specifies the gaseous streams or effluent release points that should be monitored and sampled. In addition, for monitoring the effluents during a postulated event, Section 11.5 of the SRP[11] states that "Provisions should be made for monitoring instrumentation, sampling, and sample analyses for all identified gaseous effluent release paths in the event of a postulated accident."

As explained earlier, implementation of the proposed solutions has no impact on the core-melt accident frequency. Moreover, "while protective actions can be recommended based on effluent releases in progress, the probability for a core-melt scenario was such that actions would be recommended based on anticipated releases prior to the actual release themselves. Under this assumption, monitoring effluent releases would have little or no impact on public risk and would be mainly for confirmation and quantification."

Specific requirements related to some of the factors in the proposed design criteria mentioned in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," have not been established; however, based on the review of the NRC's regulations presented above, the staff concluded that the overall objectives of Item III.D.2.1(1) are met by the existing regulations. Moreover, the low safety significance of the issue does not warrant further actions to evaluate and implement the proposed solutions. Therefore, the staff changed the status of Item III.D.2.1(1) and DROPPED this item from further pursuit.[1964]

ITEM III.D.2.1(2): STUDY THE FEASIBILITY OF REQUIRING THE DEVELOPMENT OF EFFECTIVE MEANS FOR MONITORING AND SAMPLING NOBLE GASES AND RADIOIODINE RELEASED TO THE ATMOSPHERE

In addition to Criterion 64 of Appendix A to 10 CFR Part 50, the regulation at 10 CFR 50.34(f)(2)(xvii) establishes the requirement for monitoring noble gas effluents and continuous sampling of radioactive iodines and particulates in gaseous effluents. According to this part of the regulation, “each applicant for a light-water reactor construction permit or manufacturing license whose application was pending as of February 16, 1982,” in addition to “each applicant for a design certification, design approval, combined license, or manufacturing license under part 52” of 10 CFR, needs to “Provide instrumentation to measure, record and readout in the control room:…(E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.”

Based on the review of the NRC regulations related to this issue presented above and the low safety significance of this issue, the staff concluded that Item III.D.2.1(2) is adequately addressed by the existing regulations. Therefore, the staff changed the status of Item III.D.2.1(2) and DROPPED this item from further pursuit.[1964]

ITEM III.D.2.1(3): REVISE REGULATORY GUIDES

NUREG-0660[48] called for this issue to “revise Regulatory Guide 1.21, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, Standard Review Plan Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems, and further revise Regulatory Guide 1.97, as necessary.” All of these documents have been updated since the issuance of NUREG-0660.[48] Some specific factors of the design criteria mentioned in NUREG-0660[48] have not been included in these updates. However, the overall objective of the issue has been thoroughly addressed in these updates. As of April 2010, the latest revision of each document is available as follows: Regulatory Guide (RG) 1.21, Revision 2, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” issued June 2009[1968]; SRP[11] Section 11.5, issued March 2007; and RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” Revision 4, issued June 2006.[55]

Because of the revisions made to RG 1.21,[1968] SRP[11] Section 11.5, and RG 1.97,[55] the staff changed the status of Item III.D.2.1(3) and DROPPED this item from further pursuit.[1964]

ITEM III.D.2.2: RADIOIODINE, CARBON-14, AND TRITIUM PATHWAY DOSE ANALYSIS

The four parts of this item were combined and evaluated together.

Description

Historical Background

This TMI Action Plan[48] item addressed the issue of further research for improving the understanding of radioiodine partitioning in nuclear power reactors and of the environmental behavior of radioiodine, carbon-14, and tritium following an accident and during normal operation.

Iodine isotopes are considered to be major contributors to the occupational and public dose during a loss-of-coolant accident, along with noble gases and fission products. A study in these areas was documented in NUREG-0772, "Technical Bases for Estimating Fission Product Behavior during LWR Accidents," issued June 1981,[212] with the following major conclusions: (1) uncertainties in predicting atmospheric release source terms were very large (at least a factor of 10), (2) source terms for certain accident sequences may have been overestimated in past studies; e.g., WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," issued October 1975,[16] and (3) cesium iodide should be the predominant chemical form of iodine under severe accident conditions.

Safety Significance

The above conclusions indicated that the methodology and assumptions used for evaluating radioiodine release could result in unrealistic estimates (e.g., RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,"[213] and RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"[214]). Also indicated was that more research in aerosol behavior and fission product chemistry was needed in order to improve and support the calculation methodology concerned with radioiodine partitioning, fission product behavior, and related topics.

Possible Solution

The NRC assumed that further study would improve the understanding of this issue and result in more realistic assumptions and methods for evaluating source terms, releases, and the environmental behavior of radioiodine, carbon-14, and tritium following an accident. This research would not affect accident frequencies at nuclear power plants. However, the NRC assumed that the results of these studies would be used to revise the SRP[11] and RGs.

The NRC also assumed that the RG revisions could result in reducing the size of existing emergency planning zones from a 10-mile radius to a 2-mile radius. This assumption was based on a reduction of source terms in a core-melt accident by a factor of 10. This would result in reducing dose concentration at a particular distance from the nuclear reactor also by a factor of 10. Assuming neutral weather conditions with a 30-meter-high plume, the offsite dose predicted at 2 miles from the accident scene, using the reduced source term assumption, would be the same as that predicted at 10 miles from the reactor.

Conclusion

The study of radioiodine, carbon-14, and tritium behavior at Three Mile Island Unit 2 (TMI-2) called for in Item III.D.2.2(1) was completed in June 1981 and documented in NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions," issued June 1981,[455] and NUREG-0772.[212] Items III.D.2.2(2), (3), and (4) called for a series of studies and evaluations of various radionuclide pathways and models followed, if necessary, by revisions to several SRP[11] sections and RGs. As part of the staff's task to prepare and publish a manual (referred to as the "Offsite Dose Calculation Manual"[598]) to be used by the NRC and industry to estimate individual and population doses during normal and accident conditions, Items III.D.2.2(2), (3), and (4) were assessed. This Offsite Dose Calculation Manual was prepared under Item III.D.2.5 and fully described each of the theoretical models used to predict radionuclide transport.[149] Thus, Items III.D.2.2(2), (3), and (4) were covered under Item III.D.2.5.

ITEM III.D.2.2(1): PERFORM STUDY OF RADIOIODINE, CARBON-14, AND TRITIUM BEHAVIOR

This item was evaluated in Item III.D.2.2 above and was RESOLVED with no new requirements.

ITEM III.D.2.2(2): EVALUATE DATA COLLECTED AT QUAD CITIES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.2(3): DETERMINE THE DISTRIBUTION OF THE CHEMICAL SPECIES OF RADIOIODINE IN AIR-WATER-STEAM MIXTURES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.2(4): REVISE SRP AND REGULATORY GUIDES

This item was evaluated in Item III.D.2.2 above and was determined to be covered in Item III.D.2.5.

ITEM III.D.2.3: LIQUID PATHWAY RADIOLOGICAL CONTROL

The four parts of this item were combined and evaluated together.

Description

This TMI Action Plan[48] item was concerned with improving public radiation protection in the event of a nuclear power plant accident by improving the control of radioactivity released into the liquid pathway. This control could be accomplished by the application of various interdictive measures at the source of the release and/or along the liquid pathway. Techniques were developed and were being used to evaluate the liquid pathway effects of an accident for each reactor site. Sites that might require interdictive measures related to liquid pathway releases were to be determined. Interdictive measures were to be assessed as to their effectiveness in improving public radiation protection.

Conclusion

A liquid pathway analysis for Zion Nuclear Power Station was completed by the Office of Nuclear Reactor Regulation's Division of Engineering in 1980.[391] In addition, a liquid pathway analysis was performed for the Indian Point nuclear power plant. Both analyses were used in NUREG-0850, “Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects,” issued November 1981.[390] A simplified analysis for liquid pathway studies (NUREG-1054, “Simplified Analysis for Liquid Pathway Studies,”)[658] was published in August 1984, and Section 7.1.1 of NUREG-0555, “Environmental Standard Review Plans for the Environmental Review of Construction Permit Applications for Nuclear Power Plants” (the ESRP), issued May 1979,[464] was drafted with no new requirements for licensees or applicants.[659,660] ESRP Section 7.1.1 was finally published as NUREG-1165, “Environmental Standard Review Plan for ES Section 7.1.1,”[838] in November 1985. Thus, this item was RESOLVED and no new requirements were established.[799]

ITEM III.D.2.3(1): DEVELOP PROCEDURES TO DISCRIMINATE BETWEEN SITES/PLANTS

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.3(2): DISCRIMINATE BETWEEN SITES AND PLANTS THAT REQUIRE CONSIDERATION OF LIQUID PATHWAY INTERDICTION TECHNIQUES

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.3(3): ESTABLISH FEASIBLE METHOD OF PATHWAY INTERDICTION

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.3(4): PREPARE A SUMMARY ASSESSMENT

This item was evaluated in Item III.D.2.3 above and was RESOLVED with no new requirements.[799]

ITEM III.D.2.4: OFFSITE DOSE MEASUREMENTS

ITEM III.D.2.4(1): STUDY FEASIBILITY OF ENVIRONMENTAL MONITORS

Description

This TMI Action Plan[48] item called for the staff to study the feasibility of environmental monitors capable of measuring real-time rates of exposure to noble gases and radioiodines. Monitors or samplers capable of measuring respirable concentrations of radionuclides and particulates were also considered. This activity supported proposed revisions to RG 1.97[55] (see Item II.F.3).

Conclusion

The establishment of guidance in RG 1.97[55] for fixed monitors to detect unidentified releases was postponed pending the outcome of a feasibility study that was completed in April 1982.[188] Using this study as a basis, the staff concluded that environmental monitors of this nature were not practical and that proposed requirements for these monitors should be dropped from consideration.[189] Thus, all required action on this item was completed[382] and the issue was RESOLVED with no new requirements.

ITEM III.D.2.4(2): PLACE 50 TLDs AROUND EACH SITE

Description

This TMI Action Plan[48] item called for Office of Inspection and Enforcement (OIE) to place 50 thermo-luminescent dosimeters (TLDs) around each site in coordination with States and utilities. During normal operation, OIE quarterly reports from these dosimeters were to be provided to NRC, State, and Federal organizations. In the event of an accident, the dosimeters could then be read at a frequency appropriate to the needs of the situation.

The specific objectives of this program were to (1) establish preoperational, historical, baseline radiation dose levels, whenever possible, for each monitored facility, (2) provide ongoing radiation dosimetry data during routine operations, (3) provide postaccident radiation dosimetry to aid in assessment of population exposures and radiological impact, (4) allow for independent verification of the adequacy of NRC licensees' environmental radiation monitoring programs, (5) provide uniform treatment of dosimeters with respect to handling, shipping, calibrating, reading, and data processing for all monitored facilities in the United States, and (6) provide uniform, consistent environmental radiation monitoring data for use by the Congress, Federal and State agencies, monitored facilities, and the public.

This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered to be a licensing issue.

Conclusion

OIE completed installation of TLDs at all operating reactors in August 1980 in accordance with the TMI Action Plan schedule. A direct radiation monitoring network was established and a program for routine reporting began. The completion of these activities was described in an OIE memorandum.[236] With the establishment of the NRC TLD direct radiation monitoring network, the installation of TLDs at all operating reactor sites, and the routine reporting of the TLD measurements, all work required by this item was completed.[236,379] Thus, this licensing issue was resolved.

ITEM III.D.2.5: OFFSITE DOSE CALCULATION MANUAL

Description

Historical Background

This TMI Action Plan[48] item called for the Office of Nuclear Reactor Regulation to prepare a manual to be used by the NRC and plant personnel to estimate the maximum individual doses and population doses during an accident.

Safety Significance

This issue did not affect core-melt frequency or the amount of radioactivity released. Instead, it was intended to reduce the consequences of a major release by assuring that licensees have a rapid and sufficiently accurate method of estimating dose, and that communication between licensees and the NRC be expedited by a common standard calculation method used by both.

Possible Solution

The proposed manual was expected to include formulations with which to combine source term and meteorological measurements. This would determine offsite dose rates in a manner that would be standard among all parties making decisions on public protection and emergency response. Appendix 2 to NUREG-0654[224] established criteria for automated assessment of radiation doses in the event of an accident.

Priority Determination

Frequency Estimate

Because the proposed solution to the issue did not affect core-melt accident frequency, the frequencies for the various release categories given for Oconee Nuclear Station, Unit 3, and Grand Gulf Nuclear Station, Unit 1, were used unchanged in the value/impact calculation.

Consequence Estimate

In an assessment[64] of this issue, PNL experts judged that a 1-percent reduction in public dose (man-rem) might be expected as a result of having an offsite dose calculation manual available. It was estimated that the changes in consequences would be much less (0.01 percent to 0.1 percent). Because all sequences would be affected and the risk from both PWRs and BWRs was about 210 to 250 man-rem/reactor-year (RY), the risk reduction was estimated to be 0.02 to 0.2 man-rem/RY.

At the time of the evaluation of this issue in November 1983, there were 43 PWRs and 27 BWRs operating, with cumulative experience of 350 RY and 260 RY, respectively. Considering the 36 PWRs and 21 BWRs that were under construction and assuming a plant life of 40 years, there were 2,810 PWR-years and 1,660 BWR-years in the future, for a total of 4,470 RY. Therefore, the total risk reduction associated with this issue was (0.2)(4,470)man-rem or 894 man-rem.

Cost Estimate

Industry Cost: For licensees, 4 man-weeks of training for implementation were assumed, since operators were being retrained periodically and this retraining could include dose calculation methods. This different method would not incur additional recurring costs. Thus, the total industry cost was estimated to be $7,700/plant or $0.98M for 127 plants.

NRC Cost: The NRC had already completed work on development of a portable computerized system for dose calculations to be used by the NRC regional offices. This was part of the program for NUREG-0654.[224] This program was developed to the point of field trials for the computerized system. Based on the development costs, an additional $125,000 to develop this package into a manual form for use by utilities was assumed. It was estimated that NRC site representatives could spend a minimal amount of time (about 2 days) to evaluate initial utility performance with the package. This was estimated to be $600/plant. Thus, the total NRC cost was approximately $200,000 for all plants.

Total Cost: The total industry and NRC cost associated with the possible solution was $(0.98 + 0.2)M or approximately $1.2M.

Value/Impact Assessment

Based on an estimated public risk reduction of 894 man-rem and a cost of $1.2M for a possible solution, the value/impact score was given by the following:

$$S = \frac{894 \text{ man-rem}}{\$1.2\text{M}}$$

$$= 758 \text{ man-rem}/\$\text{M}$$

Conclusion

Based on the above value/impact score, the issue would have had a MEDIUM priority ranking (see Appendix C). However, before approval of the prioritization evaluation in November 1983, the Offsite Dose Calculation Manual was published as NUREG/CR-3332, “Radiological Assessment—A Textbook on Environmental Dose Analysis,”[599] in September 1983. Thus, the issue was RESOLVED and no new requirements were issued.[598]

ITEM III.D.2.6: INDEPENDENT RADIOLOGICAL MEASUREMENTS

Description

This TMI Action Plan[48] item dealt with independent radiological measurements; i.e., means of collecting data independent of licensees’ programs. An OIE task force developed a plan and requirements for upgrading the capability of regional offices to perform independent radiological measurements during routine inspections and emergency response operations. The objective of the upgrade was to achieve consistent capability among the regional offices, including standardization in major equipment items such as mobile laboratory vans, gamma spectrum analysis equipment, radiation survey instrumentation, and air-sampling and monitoring devices.

Based on the recommendations of the task force, each region was equipped with complete mobile laboratories.[235] In some cases, this represented upgrading certain equipment or purchasing new equipment. This action item required that revisions be made to the inspection program to include the upgrading of the independent radiological measurements. The program was included in the routine OIE program for review and revision of the inspection program. As new equipment needs were identified, the program was to be revised and the equipment acquired.

This item addressed improvements in the NRC capability to make independent assessments of safety and, therefore, was considered to be a licensing issue.

Conclusion

With the upgrading of independent radiological measurements and the implementation of other recommendations made by the task force, all work required by this item was completed.[235,379] Thus, this licensing issue was resolved.

References

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16. WASH-1400 (NUREG-75/014), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Atomic Energy Commission, October 1975.

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

55. Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, December 1975, (Rev. 1) August 1977, (Rev. 2) December 1980, (Rev. 3) May 1983, (Rev.4) June 2006.

57. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," U.S. Nuclear Regulatory Commission, July 1979.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

149. Memorandum for J. Funches from R. Mattson, "Comments on Prioritization of Licensing Improvement Issues," February 2, 1983. [8401170099]

188. NUREG/CR-2644, "An Assessment of Offsite, Real-Time Dose Measurements for Emergency Situations," U.S. Nuclear Regulatory Commission, April 1982.

189. Memorandum for K. Goller from R. Mattson, "Proposed Changes to Regulatory Guide 1.97," July 29, 1982. [8208060339]

212. NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," U.S. Nuclear Regulatory Commission, June 1981.

213. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," U.S. Nuclear Regulatory Commission, November 1970, (Rev. 1) June 1973, (Rev. 2) June 1974. [7907100054]

214. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, November 1970, (Rev. 1) June 1973, (Rev. 2) June 1974. [7907100058]

224. NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, February 1980, (Rev. 1) November 1980.

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TASK IV.E: SAFETY DECISION-MAKING

The objective of this task is to develop plans for an integrated program of safety decision-making. These plans include: (1) an expanded program of regulatory research covering methodologies for making safety decisions and safety-cost tradeoffs, with application both to decisions regarding the overall risk of nuclear power plants and the nuclear fuel cycle and to specific licensing and inspection decisions; (2) early resolution of safety issues after they are identified, including application of the decisions to operating reactors, reactors under construction, and standard designs; (3) elimination of repetitive consideration of identical issues at several stages of the licensing process; (4) expanded use of rulemaking to implement safety criteria developed as a result of the various Task Action Plans; and (5) improved and expanded systematic assessments of operating reactors.

ITEM IV.E.1: EXPAND RESEARCH ON QUANTIFICATION OF SAFETY DECISION-MAKING

Description

This issue is described in NUREG-0660[48] as follows:

"The purpose of this task is to proceed toward better quantification of safety objectives, including safety-cost tradeoffs. The concept will use ongoing research that one might quantify risk and possible application of formal decision-making techniques to the regulatory environment. Future programs will build on the risk assessment and systems reliability work currently underway and incorporate a better assessment of common-mode and human failures. Safety objectives will be developed for components and systems, and eventually these might be amalgamated into a more tightly bounded, quantitative safety standard, as opposed to a safety objective having fairly large inherent uncertainties."

The approach to the resolution of this item is also outlined in NUREG-0660[48] as follows:

(1) RES will assemble a research task force from a wide variety of professional disciplines. The task force will formulate several possible sets of numerical criteria using different technical approaches. The formation of the research task force and the conduct of its meetings are being coordinated through IEEE with cooperation from other professional societies.

(2) BNL has been contracted to independently formulate criteria to investigate the implications of safety criteria and to determine the impact of attempting to satisfy such criteria.

(3) Decision theory and survey methods for obtaining criteria are being investigated as extensions of previous projects on risk analysis. These methods can provide a separate approach to obtain acceptable risk criteria.

(4) Negotiations are underway with various governmental and private agencies for information on proposed criteria. In addition, letters have been sent to several hundred individuals announcing the project and requesting their contributions.

(5) To assure that the criteria receive rigorous peer review, negotiations are underway with the National Science Foundation, the National Academy of Sciences, and the American Statistical Association.

The current accomplishments include completion of NUREG/CR-1614,[275] NUREG/CR-1539,[276] NUREG/CR-1930,[277] NUREG/CR-1916,[278] and NUREG/CR-2040.[279] The current status is such that PNL, ORNL, BNL, ANL, IEEE, NRC (Office of Policy Evaluation), and the ACRS are completing various elements of the overall program. These activities will develop and exhibit approaches with which to better factor risk evaluation into NRC decision-making regarding reactor plant safety. This issue does not appear to have a direct effect on public risk reduction or to have any industry cost directly associated with its resolution. Therefore, it is a licensing issue.

Conclusion

A value/impact handbook (NUREG/CR-3568)[970] was developed by the staff to support specific cost/benefit analyses of proposed rules. In November 1986, RES determined that all other staff work required by this issue was being pursued in the ongoing work related to the Commission's Safety Goal.[954] Thus, this Licensing Issue has been resolved.

ITEM IV.E.2: PLAN FOR EARLY RESOLUTION OF SAFETY ISSUES

Description

This TMI Action Plan[48] item required NRR, in consultation with other appropriate offices, to develop a plan for the early identification, assessment, and resolution of safety issues. This item is related to the establishment and implementation of an NRC program to identify and resolve safety issues and, therefore, is considered a licensing issue.

Conclusion

The plan was presented in SECY-81-513[1] on August 25, 1981 and is currently being implemented by SPEB. Thus, this Licensing Issue has been resolved.

ITEM IV.E.3: PLAN FOR RESOLVING ISSUES AT THE CP STAGE

Description

According to NUREG-0660,[48] NRR and ELD transmitted a consent calendar item to the Commission on February 14, 1980, entitled "Response to Staff Requirements Memorandum (Affirmation Session 79-40) With Respect to Post-CP Design and Other Changes," SECY-80-90. This paper discussed five options regarding the establishment of construction requirements. The recommendation of this consent paper is to publish an advance notice of public rulemaking to obtain comments on these options. After receipt of public comment on the above, the staff will prepare a plan to implement methods to resolve as many issues as possible at the construction permit stage before major financial commitments in construction occur.

An advanced notice of rulemaking was published in the Federal Register in December 1980 with a public comment period ending on February 9, 1981. On August 18, 1981, the Director of the Division of Risk Analysis sent a memo to distribution proposing an approach to the Rule and requested examples of the types of characteristic alterations representing post-CP changes. The draft of the Rule is currently being reviewed.

In view of the intent of this item, it is concluded that its resolution does not have a direct effect on public risk reduction and is, therefore, considered to be a licensing issue.

Conclusion

Staff stated in the Supplement to this report published in 1986 that the resolution of this Licensing Issue was available. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM IV.E.4: RESOLVE GENERIC ISSUES BY RULEMAKING

Description

This TMI Action Plan[48] item states that the NRC will undertake the additional task of developing a program for reviewing new criteria before their promulgation to determine whether rulemaking would be the desirable means of implementation. The intent will be to implement new NRC criteria by rule, wherever feasible and timely, instead of by license changes, orders, or changes in regulatory guides.

This item does not have a direct effect on public risk reduction nor is there any industry cost associated with the completion or implementation of the issue resolution. Thus, it is considered a licensing issue.

Conclusion

In November 1986, RES concluded that ongoing NRC activities such as the Safety Goal Program, RES independent review of rulemaking, and the Commission policy on backfitting had effectively addressed the concerns of this issue.[954] Thus, this Licensing Issue has been resolved.

ITEM IV.E.5: ASSESS CURRENTLY OPERATING REACTORS

Description

Historical Background

As part of developing plans for an integrated program of safety decision making, NRR, in consultation with other appropriate offices, will develop a plan for approval by the Commission for the systematic assessment of the safety of all operating reactors. Development of such a plan will take into account the SEP, the ACRS comments on the program, the IREP plan, and ongoing TMI lessons-learned activities. This value/impact assessment of Item IV.E.5 deals with the work under the SEP. Value/impact assessments of IREP and NREP are presented in Items II.C.1 and II.C.2, respectively.

SEP is now reviewing the 10 oldest plants against current licensing review safety criteria, including the SRP, to provide the basis for integrated and balanced backfit decisions. This review is nearly complete and, therefore, is not part of this assessment. The next SEP phase involves evaluation of 11 additional plants. In this next phase, PRA evaluations will be coordinated with the deterministic review method (review against current licensing safety criteria). The PRA will be done as part of NREP (TMI Action Plan Item II.C.2).

Possible Solutions

As safety-related problems are identified for each plant, resolutions are developed using procedural and administrative changes, possible credit for non-safety systems where justified, and hardware backfits as necessary to reduce risk levels. The process used to decide appropriate corrective actions employs the judgment of a team of NRC staff familiar with each plant.

Priority Determination

This priority determination uses potential risk reduction analyses and cost estimate information provided by PNL.[64]

Frequency/Consequence Estimate

This public risk reduction analysis for SEP considers only the 11 additional plants currently proposed to be reviewed in the first group of Phase III plants, since much of the review of the first 10 plants in Phase II has been performed. The 11 plants consist of 7 PWRs and 4 BWRs with estimated average remaining lives of 24 and 22 years, respectively. In Item II.C.2 (NREP), it is estimated that an overall core-melt frequency reduction of 2×10^{-4}/RY could be achieved for one-third of the plants to be reviewed under NREP. Although the NREP evaluation of these plants will identify some areas of potentially high risk, the NREP methods do not address areas such as external events and structural design which are included in the SEP deterministic review. For this issue, it is assumed that the risk reduction estimated for NREP could be achieved by the SEP considering only the benefit resulting from using the deterministic review method for external events and other issues outside the scope of PRAs (e.g., adequacy of design, structural issues, and design errors).

Using the base case core-melt frequency and the base case public risk for each type plant, and assuming a population of 340 persons per square mile over an area having a 50 mile radius, the average risk per core-melt is 2.5×10^{6} man-rem for PWRs and 6.8×10^{6} man-rem for BWRs.

Using the average risk value and the assumption stated above that the deterministic review method can achieve the core-melt frequency reduction estimated for NREP for one-third of the plants reviewed, we can estimate the potential reduction for the SEP Phase III as follows:

<u>PWRs</u>: Risk Reduction = (2.5×10^{6} man-rem/core-melt)(2×10^{-4} core-melt/RY)

$= 500$ man-rem/RY

BWRs: Risk Reduction = (6.8×10^{6} man-rem/core-melt)(2×10^{-4} core-melt/RY)
= 1,360 man-rem/RY

Summed over the average remaining plant life for the 11 plants proposed, the total public risk reduction is calculated to be approximately 80,000 man-rem.

Cost Estimate

Industry Cost: Based on SEP studies completed to date, the following costs per plant are estimated: up to $2M for engineering studies to identify areas of plant modification and $2M to $20M to design and install modifications.

For purposes of this analysis, assume a conservative implementation cost per plant of $2M for engineering studies at each of the 11 plants plus $10M average design and installation (including capital equipment cost) at one-third of the plants. For 11 plants, the total industry cost is $[(11)(2) + (1/3)(11)(10)]M or $55M.

NRC Cost: Based on past studies, NRC staff effort has totaled 10 man-yr/plant plus $700,000 contract technical support per plant. Thus, total development and implementation cost, at $100,000/man-year, is:

(10 man-years/plant)($100,000/man-yr) + ($700,000/plant)(11 plants) = $19M.

Assuming NRC staff effort for review and inspection of plant modifications at one-third of the plants is 0.5 man-wk/RY and the average remaining life of these plants is 23 years, then the total plant review cost is:

(0.5 man-wk/RY)($2,000/man-wk)[(1/3)(11)(23)RY] = $0.1M.

Value/Impact Assessment

Based on a public risk reduction of 80,000 man-rem, the value/impact score is given by:

$$S = \frac{80{,}000 \text{ man-rem}}{\$(55 + 19)\text{M}}$$

$$= 1{,}000 \text{ man-rem/\$M}$$

Other Considerations

If the cleanup of an accident is assumed to require 19,900 man-rem and the same assumption on accident frequency reduction is retained, the total reduction in occupational exposure would be 170 man-rem. An estimate of the occupational exposure to implement any changes cannot be made without identifying the specific changes. However, there would likely be some increase in occupational exposure, but it would be small compared to the public risk reduction.

An additional consideration is that plant damage is estimated to be $1,650M per plant for core-melt. Thus, total averted plant damage for one-third of the plants with a reduced core-melt frequency could be

($1,650M)(2 x 10^{-4}/RY)[(1/3)(11)(23)RY] = $28.9M

Uncertainties

Since the 11 plants considered are older plants, it is possible that the assumed 10^{-4}/RY risk reduction may be achieved for more than one-third of the 11 plants as assumed, thus resulting in greater risk reduction with an associated increase in implementation cost. However, the value/impact score would not change appreciably.

Conclusion

The value/impact score indicates a medium priority. However, the potentially large, though uncertain, risk reduction of nearly 80,000 man-rem justified a high priority ranking.

Work completed by the staff on this item was closely related to the accomplishments under Item II.C.2. Whereas Item II.C.2 called for the initiation of IREP studies (i.e. plant-specific PRAs) on all remaining operating reactors, Item IV.E.5 called for the development of a plan for the systematic assessment of the safety of all operating reactors. The Integrated Safety Assessment Program (ISAP), presented in SECY-84-133[814] and SECY-85-160,[815] provided for a comprehensive review of selected operating reactors to address all pertinent safety issues and to provide an integrated cost-effective implementation plan for making needed changes. Under ISAP, each plant would be subject to an integrated assessment of safety topics, a probabilistic safety assessment, and an evaluation of operating experience.

NRC guidance, as described in the Severe Accident Policy Statement (see Item II.B.8), states that OLs will be expected to perform plant-specific PRAs in order to discover instances of particular vulnerability to a core-melt or poor containment performance, given a core-melt. Thus, this item was RESOLVED and no new requirements were established.[816]

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275. NUREG/CR-1614, "Approaches to Acceptable Risk: A Critical Guide," U.S. Nuclear Regulatory Commission, September 1980.

276. NUREG/CR-1539, “A Methodology and a Preliminary Data Base for Examining the Health Risks of Electricity Generation from Uranium and Coal Fuels,” U.S. Nuclear Regulatory Commission, August 1980.

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278. NUREG/CR-1916, "A Risk Comparison," U.S. Nuclear Regulatory Commission, February 1981.

279. NUREG/CR-2040, “A Study of the Implications of Applying Quantitative Risk Criteria in the Licensing of Nuclear Power Plants in the U.S.,” U.S. Nuclear Regulatory Commission, March 1981.

814. SECY-84-133, “Integrated Safety Assessment Program (ISAP),” U.S. Nuclear Regulatory Commission, March 23, 1984. [8404100072]

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816. Memorandum for W. Dircks from H. Denton, “Close-out of Generic Issues II.C.2, ‘Continuation of IREP,’ and IV.E.5, ‘Assess Currently Operating Reactors,’” September 25, 1985. [9909290069]

954. Memorandum for V. Stello from E. Beckjord, “Closeout of TMI Action Plan Items,” November 13, 1986.

970. NUREG/CR-3568, “A Handbook for Value-Impact Assessment,” U.S. Nuclear Regulatory Commission, December 1983.

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

TASK V.D: LICENSING PROCESS

The objective of this task was to enhance public participation in, and make needed reforms to, the nuclear licensing process.

ITEM V.D.1: IMPROVE PUBLIC AND INTERVENOR PARTICIPATION IN THE HEARING PROCESS

Description

This NUREG-0660,[48] Rev. 1 item called for the Commission to assess alternative methods to enhance public and intervenor participation in the hearing process by undertaking a pilot program for intervenor funding in accordance with the FY-81 budget request and by studying the concept of an Office of Hearing Counsel, as described by the President's Commission recommendation, and other concepts of Public Counsel (such as the Office of Public Counsel recommended by the NRC Special Inquiry Group or concepts used by some Public Service Commissions). If such concepts proved to be desirable, the Commission was to propose the needed legislation. This item was originally identified as Item 5 in Chapter V but was made Item V.D.1 in Rev. 1 to NUREG-0660.[48]

The NRC sought authorization to establish a pilot program[872] to fund intervenors in its budget request for FY-81. Congress not only failed to enact such legislation, but included a provision in NRC's 1981 Appropriations Act (Public Law 96-367)[924] which precluded the use of funds to pay the expenses of, or otherwise compensate, parties intervening in NRC proceedings. After enactment of this legislation and issuance of an opinion by the Comptroller General on December 3, 1980, the NRC terminated[873,874] a one-year pilot program it had established to provide intervenors certain forms of procedural assistance, such as free hearing transcripts and service of documents. Congress also barred the NRC from funding intervenors in FY-82 and FY-83. Prior to Congressional action, OGC had begun a review of the desirability of creating an Office of Public Counsel. After Congress prohibited intervenor funding, OGC terminated its review.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence is assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

This Licensing Issue has been resolved.

ITEM V.D.2: STUDY CONSTRUCTION-DURING-ADJUDICATION RULES

Description

This NUREG-0660,[48] Rev. 1 item called for the Commission to complete rulemaking on whether construction should be permitted while challenges to a construction permit authorized by a licensing board are under administrative adjudication. This item was originally identified as Item 6 in Chapter V but was made Item V.D.2 in Rev. 1 to NUREG-0660.[48]

Following the TMI-2 accident, the Commission suspended in part its so-called immediate effectiveness rule. This rule had authorized the issuance of reactor construction permits or operating licenses immediately upon receipt of a favorable licensing Board decision, notwithstanding the filing of administrative appeals. In its place, the Commission instituted a mandatory review procedure for such decisions. In 1981, the rule was partially reinstated with respect to decisions authorizing the issuance of a reactor operating license. The rule, as applied to decisions authorizing reactor construction, has been the subject of a separate rulemaking.

The Commission published a notice of proposed rulemaking and requested comments on several options for amending the immediate effectiveness rule for construction permit decisions.[875] On October 25, 1982, the Commission published a proposed rule that would make the effectiveness review procedures for construction permits conform to those for operating licenses.[876] The Commission noted that it was still considering the various options presented and that revisions might be proposed later as part of broader reforms to the Commission's hearing process. As a result of further consideration, the Commission now has pending before it a new rulemaking proposal relative to immediate effectiveness reviews for both construction permits and operating licenses.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

Staff stated in the Supplement to this report published in 1986 that a solution to this Licensing Issue was available, but the item had not been resolved. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM V.D.3: REEXAMINE COMMISSION ROLE IN ADJUDICATION

Description

This NUREG-0660[48] Rev. 1 item called for the Commission to review its role in adjudications to examine the extent of Commission involvement in licensing proceedings and to eliminate any undesirable and unnecessary insulation of the Commission from decision-making activities of the staff. This item was originally identified as Item 17 in Chapter V but was made Item V.D.3 in Rev. 1 to NUREG-0660.[48]

The Commission's role in adjudication is addressed under three topics: the immediate effectiveness review, the appellate process structure, and communications between the Commission and the staff.

Immediate Effectiveness Reviews: Following the TMI-2 accident, the Commission promulgated amendments to its regulations (10 CFR 2.764) which increased the Commission's role in adjudications. Under the revised regulations, decisions by Atomic Safety and Licensing Boards (ASLB) which authorize a utility to operate a facility at full power do not become effective upon issuance. Instead, the Commission conducts an "immediate effectiveness review" to determine whether the ASLB decision should be effective during the pendency of administrative appeals. The Commission seeks to complete these reviews within 30 days of the ASLB decision, or on an otherwise timely basis when the applicant has not completed construction or is not otherwise ready to operate at full power.

In 1981, the Commission established a Regulatory Reform Task Force to examine the NRC's licensing process. This Task Force recommended a different approach; it advocated the "immediate effectiveness" rule that was in place prior to the TMI-2 accident, i.e., construction permits and operating licenses should be issued on the basis of favorable ASLB decisions with an immediate effectiveness review by the Commission. In October 1982, the Commission issued for public comment a Notice of Proposed Rulemaking[876] which, if adopted, would extend the immediate effectiveness review procedures to ASLB decisions which authorize the issuance of construction permits or limited work authorizations.

As is indicated in the discussion under Item V.D.2 above, the Commission now has pending before it a new rulemaking proposal relative to immediate effectiveness reviews for operating licenses.

Structure of the Appellate Process: The Commission has a three-tier adjudicatory system. Matters are first heard by an ASLB, followed in most cases by a mandatory review by an ASLAB and then by a discretionary Commission review. In December 1979, OGC prepared a study of the Commission's appellate system. One option examined, but not recommended, was to increase the Commission's adjudicatory role by eliminating the ASLAB. After receiving public comments on the study, the Commission decided not to abolish ASLAB review. The Regulatory Reform Task Force recommended to the Commission that it remove the ASLAB as an intermediate appeal body, but assign it responsibility of drafting Commission adjudicatory orders. The Commission did not adopt this recommendation.[984]

Communications Between the Commission and the Staff: The Commission's Regulatory Reform Task Force recommended that the Commission modify its separation of functions (10 CFR 2.719) and ex parte rules (10 CFR 2.780) to permit greater communication between the Commission and the staff on matters under adjudication.

On March 26, 1986, the Commission published a proposed rule to revise the Commission's separation of functions and ex parte rules.[877] Present rules preclude communications between the Commission and any NRC staff member concerning a substantive matter at issue in a formal adjudicatory proceeding. Under this proposed rule, only those members of the NRC staff who are involved in an "investigative or litigative" function relative to a particular proceeding would be barred from communicating with the Commission on disputed issues in the proceeding, thereby allowing for much wider Commission access to staff expertise.

On November 2, 1983, the Commission published in the Federal Register an Advanced Notice of Proposed Rulemaking on the role of the NRC staff in the licensing process.[985] After evaluating the public comments, the Commission determined that no change should be made in the staff's role and accordingly withdrew its Advance Notice of Proposed Rulemaking.[986]

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

Staff stated in the Supplement to this report published in 1986 that a portion of this item had not been resolved. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issuess are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM V.D.4: STUDY THE REFORM OF THE LICENSING PROCESS

Description

This NUREG-0660,[48] Rev. 1 item called for the Commission to study alternatives to reform the licensing process. One suggested reform would abolish the present two-step process for initial licensing and would substitute a one-step process with increased public involvement prior to the hearing. It would also involve continued NRC jurisdiction after issuance of the single permit to verify that plant construction conforms with plans and permit specifications. The Commission was to study the standardization of nuclear power plants and consider suspending review and proceedings for applications for CPs and LWAs until the reform issues were resolved. This item was originally identified as Item 9 in Chapter V but was made Item V.D.4 in Rev. 1 to NUREG-0660.[48]

In its first formal response to the President's Commission on the TMI-2 accident, the Commission noted that a revision of licensing procedures to emphasize early and effective resolution of safety issues would require legislation (NUREG-0632).[878] Prior to forwarding proposed legislation to the Congress, the Commission took steps to improve the balance and efficiency of the power reactor licensing process. In May 1981, a statement of policy on the conduct of licensing proceedings was issued describing procedural devices which could expedite the hearings and providing Commission guidance on the use of such tools.[879] In addition, the Commission's rules of practice (10 CFR 2) were amended to expedite certain aspects of adjudicatory proceedings. Two rules were promulgated in 1982: (1) elimination of the need for power and alternative energy source issues from reactor operating license proceedings; and (2) elimination of the requirements for the review of financial qualifications of state-regulated public utilities applying for permits or licenses. Both of these rules were expected to further expedite licensing hearings. In view of the limitations of rulemaking as a means of reforming the nuclear power plant licensing process, the Commission proceeded to develop proposals for statutory changes that would accomplish the desired reforms.

In November 1981, the Commission established the Regulatory Reform Task Force[880] to review the NRC's licensing process. As a result of the efforts of this group and senior NRC officials, the Commission in June 1982 issued for public comment a draft of proposed legislation, "Nuclear Standardization Act of 1982," which included provisions for one-step licensing, issuance of a combined construction permit and operating license, and licensing of standardized plant. After review and consideration of the public comments and comments provided by an Ad Hoc Committee for the Review of Nuclear Reactor Licensing Reform Proposals, the Commission developed a draft bill, "Nuclear Power Reactor Licensing Reform Act of 1983," and on February 21, 1983 forwarded it to the Congress.[881] The 98th Congress did not act on the Commission's 1983 legislative proposal. The Commission submitted a revised proposal to the 99th Congress in 1985, but again Congress did not act.

The Regulatory Reform Task Force proposed that a number of reforms be accomplished via rulemaking: (1) amendment of 10 CFR 50 to modify the backfitting provision and associated sections applicable to reactors; (2) amendment of 10 CFR 2 and 10 CFR 50 to improve the quality of the hearing process; (3) amendment of 10 CFR 2 regarding separation of functions and ex parte communications in on-the-record adjudications; and (4) amendment of 10 CFR 2 to limit NRC staff participation as a party in contested initial license proceedings to issues on which the staff disagrees with the license applicant.

The Commission on September 20, 1983 issued a policy statement[882] on revising the backfitting process. It also issued an Advanced Notice of Proposed Rulemaking[883] on the backfitting process and presented a number of questions for public response. The final rule[884] on the backfitting process was published in September 1985.

The Commission on November 23, 1983 issued an Advance Notice of Proposed Rulemaking on amending its rules of practice (10 CFR 2) to change the staff's role in adjudicatory licensing hearings summarized this issue and presented a number of options for rulemaking and solicited public response to a set of questions.[877] The Commission withdrew this notice after determining that no change in the staff role was warranted.[986]

The Commission on April 12, 1984 published[987] a Federal Register notice soliciting public comments on the changes to the hearing process proposed by the Regulatory Reform Task Force. After reviewing the public comments, the Commission determined that four of the proposals merited further consideration. These were published as a proposed rule.[988] The comment on October 17, 1986 and final action on the proposals is expected in early 1987.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

Staff stated in the Supplement to this report published in 1986 that this item was only partially resolved. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

872. Federal Register Notice 45 FR 49535, "10 CFR Part 2, Procedural Assistance in Adjudicatory Licensing Proceedings," July 25, 1980.

873. Federal Register Notice 46 FR 13681, "10 CFR Part 2, Domestic Licensing Proceedings; Procedural Assistance Program," February 24, 1981.

874. Memorandum for L. Bickwit from S. Chilk, "SECY-81-391—Provision of Free Transcripts to All Full Participants in Adjudicatory Proceedings: May 11, 1981 Comptroller General Decision," February 25, 1982.

875. Federal Register Notice 45 FR 34279, "10 CFR Parts 2, 50, Possible Amendments to 'Immediate Effectiveness Rule,'" May 22, 1980.

876. Federal Register Notice 47 FR 47260, "10 CFR Part 2, Commission Review Procedures for Power Reactor Construction Permits; Immediate Effectiveness Rule," October 25, 1982.

877. Federal Register Notice 51 FR 10393, "10 CFR Parts 0 and 2, Revision of Ex Parte and Separation of Functions Rules Applicable to Formal Adjudicatory Proceedings," March 26, 1986.

878. NUREG-0632, "NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island," U.S. Nuclear Regulatory Commission, November 1979.

879. Federal Register Notice 46 FR 28533, "Statement of Policy on Conduct of Licensing Proceedings," May 27, 1981.

880. Memorandum for All Employees from N. Palladino, "Regulatory Reform Task Force," November 17, 1981.

881. Letter to the Honorable Thomas P. O'Neill, Jr. from N. Palladino, February 21, 1983.

882. Federal Register Notice 48 FR 44173, "10 CFR Part 50, Revision of Backfitting Process for Power Reactors," September 28, 1983.

883. Federal Register Notice 48 FR 44217, "10 CFR Part 50, Revision of Backfitting Process for Power Reactors," September 28, 1983.

884. Federal Register Notice 50 FR 38097, “10 CFR Parts 2 and 50, Revision of Backfitting Process for Power Reactors,” September 20, 1985.

924. SECY-96-107, “Uniform Tracking of Agency Generic Technical Issues,” U.S. Nuclear Regulatory Commission, May 14, 1996. [9605230140]

928. Memorandum for A. Thadani from T. Speis, “Generic Safety Issue (GSI)-166, ‘Adequacy of Fatigue Life of Metal Components,’” August 26, 1996. [9808210022]

984. Memorandum for J. Tourtelotte et al. from S. Chilk, “Addendum to SRM M841218—Briefing and Discussion on the Hearing Process, 2:00 p.m., Tuesday, December 18, 1984, Commissioners’ Conference Room, D.C. Office (Open to Public Attendance),” January 31, 1985. [8502060511]

985. Federal Register Notice 48 FR 50550, “10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings; Role of NRC Staff in Adjudicatory Licensing Hearings,” November 2, 1983.

986. Federal Register Notice 51 FR 36811, “10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings; Role of NRC Staff in Adjudicatory Licensing Hearings,” October 16, 1986.

987. Federal Register Notice 49 FR 14698, “10 CFR Parts 2 and 50, Request for Public Comment on Regulatory Reform Proposal Concerning the Rules of Practice, Rules for Licensing of Production and Utilization Facilities,” April 12, 1984.

988. Federal Register Notice 51 FR 24365, “10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings—Procedural Changes in the Hearing Process,” July 3, 1986.

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

TASK V.E: LEGISLATIVE NEEDS

The objective of this task was to evaluate legislative needs evidenced by and from TMI.

ITEM V.E.1: STUDY THE NEED FOR TMI-RELATED LEGISLATION

Description

This NUREG-0660,[48] Rev. 1 item called for the Commission to study the need for legislation with respect to the following:

(1) Clarification of NRC authority to issue a license amendment prior to a hearing when necessary to ensure the health and safety of the public;

(2) Determination of whether NRC should seek an amendment to the Sunshine Act to reduce the Act's requirements for Commission meetings during an emergency;

(3) Determination of NRC's current legal authority to take over and conduct cleanup actions at a nuclear facility and with respect to the Federal Government's (a) liability for damages occurring during a cleanup conducted by NRC, and (b) entitlement to reimbursement for cleanup costs;

(4) The continuing desirability of the Price-Anderson Act in two areas: (a) extraordinary nuclear occurrence, and (b) limitation of liability;

(5) Desirability of creating a new category of license to be issued in place of an operating license for a facility during an extended recovery period following a major accident;

(6) The need for new or modified NRC authority to address the establishment of a chartered national operating company or consortium.

This item was originally identified as Item 7 in Chapter V but was made Item V.E.1 in Rev. 1 to NUREG-0660.[48]

The following is a discussion of NRC actions on the six subtasks of this item:

(1) Section 12 of Public Law 97-415[928] was amended in 1983 and clarified Commission authority under Section 189a of the Atomic Energy Act of 1954. This amendment, commonly referred to as the "Sholly Amendment," clarified NRC authority to issue a license amendment prior to a hearing when necessary to ensure the health and safety of the public. Thus, this subtask was resolved.

(2) The NRC Reorganization Plan No. 1 of 1980[929] directed the Chairman to act for the Commission in emergencies. This legislation nullified the need for any amendment to the Sunshine Act which originally required Commission meetings during emergencies. Thus, this subtask was resolved.

(3) In November 1980, NRC issued NUREG-0689[931] which adddressed NRC's legal authority over cleanup activities. After receiving this document, the Commission has not sought any legislation to augment or clarify its authority. Thus, this subtask has been resolved.

(4) The Congress is expected to pass, in the next legislative session, a revision to the Price-Anderson Act which will alter the limitation on liability provisions. It is likely that the extraordinary nuclear occurrence (ENO) provisions will remain in slightly modified form. The Commission published a proposed amendment to 10 CFR 140 revising its criteria for an ENO in April of 1985.[989] A final rule is expected in 1987. This subtask will be resolved when a final rule is approved by the Commission.

(5) Although it might be convenient to have a special category of license for a facility engaged in extended recovery operations, it has been the Commission's experience with the TMI-2 cleanup phase that NRC's authority to issue orders and license amendments provides adequate flexibility for conducting recovery operations within the framework of the preexisting facility license. Accordingly, the staff determined that there was no need to develop a new license category. Thus, this subtask has been resolved.

(6) This subtask called for the formation of an industry-wide consortium to operate the nuclear plants of utilities that are unable to meet the increased regulatory demands resulting from the TMI-2 accident. The Commission has not sought legislation in this area. Thus, this subtask has been resolved.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

Staff stated in the Supplement to this report published in 1986 that this item would be resolved when Subtask (4) was completed. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

48. NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident,” U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

928. Memorandum for A. Thadani from T. Speis, “Generic Safety Issue (GSI)-166, ‘Adequacy of Fatigue Life of Metal Components,’” August 26, 1996. [9808210022]

929. Regulatory Guide 1.139, “Guidance for Residual Heat Removal,” U.S. Nuclear Regulatory Commission, May 1978.

930. NUREG-0957, “The Price-Anderson Act—The Third Decade,” U.S. Nuclear Regulatory Commission, December 1983.

931. NUREG-0689, “Potential Impact of Licensee Default on Cleanup of TMI-2,” U.S. Nuclear Regulatory Commission, November 1980.

932. SECY-83-64A, “10 CFR 140: Proposed Rule to Revise the Criteria for Determination of an Extraordinary Nuclear Occurrence,” U.S. Nuclear Regulatory Commission, August 9, 1983. [8308250291]

933. Memorandum for A. Kenneke from W. Olmstead, “Chapter 5 of TMI Action Plan,” March 16, 1984. [8404040211]

989. *Federal Register* Notice 50 FR 13978, “10 CFR Part 140, Criteria for an Extraordinary Nuclear Occurrence,” April 9, 1985.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

ITEM A-19: DIGITAL COMPUTER PROTECTION SYSTEM

Description

At the time this issue was identified in NUREG-0371,[2] trends in the design of nuclear power plants showed an increase in the use of digital computer technology in safety-related instrumentation and control systems. The first application of this technology was Arkansas Nuclear One, Unit 2 (ANO-2), where digital computers were used in the initiating logic for two reactor trip parameters. After the ANO-2 application, other digital computers, such as core protection calculators, were installed by licensees to provide reactor trip signals.

Since digital technology is considerably different from analog technology, the criteria appropriate for the safety review of digital computer-based systems are different from those used for analog-based systems. Thus, in this issue, the staff identified the need to standardize the safety review of reactor protection systems that incorporated digital computers. It was believed that the results of such standardization would be: (1) the definition of the staff's requirements for the design, development, and qualification of digital computers for use by applicants; and (2) an SRP[11] that would define uniform and consistent guidelines for the conduct of the staff's safety review.

Conclusion

In 1982, ANS and IEEE jointly approved the standard ANSI/IEEE-ANS-7-4.3.2[1324] which established a method for designing, verifying, and implementing software, and validating computer systems used in the safety-related systems of nuclear power plants.[1237] In 1985, the NRC issued Regulatory Guide 1.152[1325] which endorsed the method in ANSI/IEEE-ANS-7-4.3.2-1982.[1324] At the time this issue was evaluated in 1991, the staff was conducting a research program to investigate the use of digital computer safety systems at nuclear power plants.[1286] In particular, specific licensing needs in the area of microcomputer and Artificial Intelligence Systems had been identified and were to be addressed. The desired end product of the research effort was a regulatory guide for the design, development, acceptance testing, and periodic functional verification of Class 1E computer safety systems, and an SRP[11] addendum providing review guidance for digital computer systems in nuclear power plant safety systems (by referencing Regulatory Guide 1.152[1325] and the new regulatory guide).

Since this issue addressed the use of alternative (i.e., digital instead of analog) technology for nuclear power plant safety systems, it was not intended that the use of digital technology would result in a change in the safety of existing nuclear power plants. Thus, the issue addressed the staff's efforts in improving its capability to make independent assessments of safety and was classified as a Licensing Issue.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated

November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

11. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (1st Ed.) November 1975, (2nd Ed.) March 1980, (3rd Ed.) July 1981.

1237. NUREG/CR-5420, "Multiple System Responses Program - Identification of Concerns Related to A Number of Specific Regulatory Issues," U.S. Nuclear Regulatory Commission, October 1989.

1286. Memorandum for M. Virgilio from S. Newberry, "Proposed Research Programs to Support SICB Regulation Needs," April 26, 1990. [9005090104]

1324. ANSI/IEEE-ANS-7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," American Nuclear Society, July 6, 1982.

1325. Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, November 1985. [8511220286]

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM A-20: IMPACTS OF THE COAL FUEL CYCLE

Description

At the time this issue was identified in NUREG-0371,[2] compliance with NEPA required that alternatives to a proposed Federal action be considered, and that required alternatives be balanced against the base case in terms of their associated environmental impacts. NRC had established, through its rulemaking authority, a generic description and evaluation of the environmental impacts of the uranium fuel cycle in WASH-1248,[456] NUREG-0116,[457] and NUREG-0216.[458] Based on these studies, a summary table, Table S-3, had been prepared and promulgated as regulation in 10 CFR Part 51.20(e).

In 1978, a coal-fired plant was considered the only realistic alternative to a nuclear power plant. Existing treatment of the coal alternative was aimed essentially at economics and public health impacts; it was relatively incomplete in other areas of impact. It was believed that the comparison of the coal alternative to a proposed nuclear facility would be significantly improved, if a study were conducted for the coal fuel alternative that augmented the work that had been done by ANL in the area of health effects. Such a study would provide a comprehensive summary which evaluated the environmental effects of the coal fuel cycle in a form directly comparable to that for the uranium fuel cycle. In the absence of such a generic treatment of the effects of using coal for generating electric power, it was necessary for the staff to develop an analysis *de novo* for each licensing action, to present this individual analysis in detail in the EIS, and to defend it throughout the hearing process. It was believed that this repetitive staff effort could be avoided by preparing a generic statement suitable to support rulemaking proceedings. After the rulemaking procedure, such a statement would have the force of law necessary to avoid repetitive staff effort.

A thorough analysis of alternatives to a proposed nuclear power plant required an evaluation of the environmental effects of the coal fuel cycle to the same extent as the nuclear cycle. The environmental effects of the coal fuel cycle had long been recognized as being significant. There were deleterious effects to human health due to burning coal, but there were other significant socioeconomic and other environmental impacts at each stage of the cycle. For example, mining coal exacts a penalty in human health and safety, may require modification of large areas of land use requiring expensive reclamation and habitat restoration, and frequently produces polluting liquid and solid mine wastes. Environmental, social, economic, and health effects also accompany the transportation, storage, treatment, combustion, and waste management and disposal aspects of the fuel cycle. Failure to treat these factors had been criticized by ASLB and the ASLAB in the past, necessitating increased staff efforts in this direction.

Conclusion

This issue addressed the staff's efforts in improving its capability to make independent assessments of safety and, therefore, was considered a Licensing Issue. The issue had been covered extensively in NUREG-0252,[459] NUREG/CR-1060,[460] and NUREG-0332,[461] and further work on the subject had been discussed with personnel of the National Academy of Sciences who had expressed the view that adequate scientific bases for analyzing impacts of coal burning did not exist. It was thought that a workshop could be arranged to determine what the questions were and how they could be resolved. Definitive answers required an extensive program over a period of years and the role of the NRC in carrying out such a program was expected to be determined by the Commission.[412] The results of this issue were expected to be used in Item B-72.[2]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

2. NUREG-0371, “Task Action Plans for Generic Activities (Category A),” U.S. Nuclear Regulatory Commission, November 1978.

412. Memorandum for T. Speis from J. Funches, “Prioritization of Generic Issues—Environmental and Licensing Improvements,” February 24, 1983. [8303090540]

456. WASH-1248, “Environmental Survey of the Uranium Fuel Cycle,” U.S. Atomic Energy Commission, April 1974.

457. NUREG-0116, “Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle,” U.S. Nuclear Regulatory Commission, October 1976.

458. NUREG-0216, “Public Comments on the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle,” U.S. Nuclear Regulatory Commission, March 1977.

459. NUREG-0252, “The Environmental Effects of Using Coal for Generating Electricity,” U.S. Nuclear Regulatory Commission, June 1977.

460. NUREG/CR-1060, “Activities, Effects, and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant,” U.S. Nuclear Regulatory Commission, February 1980.

461. NUREG-0332, “Health Effects Attributable to Coal and Nuclear Fuel Cycle Alternatives,” U.S. Nuclear Regulatory Commission, November 1977.

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

ITEM A-23: CONTAINMENT LEAK TESTING

Description

Historical Background

Since the issuance of Appendix J to 10 CFR Part 50 in February 1973, certain requirements of the appendix have been found to be conflicting, impractical for implementation, or subject to a variety of interpretations by the NSSS vendors, architect-engineers, utilities, and the staff. These requirements made it difficult to determine if applicants and licensees had developed uniformly acceptable containment leak testing programs and for field inspectors to judge the acceptability of a licensee's containment leak testing practices. This also led to increases in the time devoted to leak testing that could unnecessarily delay the return of a plant to service following a refueling outage.

This NUREG-0371[2] item consists of revising and clarifying Appendix J and issuing a Regulatory Guide to establish containment leak testing methods which incorporate containment leak testing experience and are acceptable to the NRC staff. Thus far, NRR has provided recommended changes to a Working Paper[122] on Appendix J which was developed by RES in May 1982. Also, NRR is currently reviewing a draft Regulatory Guide[123] which will endorse ANSI/ANS 56.8-1981,[99] "Containment System Leakage Testing Requirements." This task was assumed not to include the issue of implementing a gross check for containment integrity which is addressed separately as TMI Action Plan Item II.E.4.3.

Safety Significance

Containment leak testing is part of assuring and verifying containment structure pressure integrity. Therefore, uncertainties in conducting containment leak tests can reduce assurance that the containment structure is being maintained to meet its design leakage rate.

Possible Solutions

Revising and clarifying the Appendix J requirements and issuing a Regulatory Guide that provides acceptable containment leak testing methods would eliminate ambiguities in the regulation, reduce the compliance burden on licensees by reducing the number of exemption requests licensees are required to submit, and reduce the paperwork burden on NRC.

Priority Determination

Frequency/Consequence Estimate

A preliminary risk reduction assessment has been performed for the containment leak rate improvements expected from the Appendix J revision and the proposed Regulatory Guide. In this assessment, we assumed that the improvement in leak testing requirements could potentially reduce the uncertainty in containment integrated leak rate test results by an amount equivalent to the design leakage rate. Improvement in containment leakage will only affect the consequence of core-melt events and mitigated LOCA events in which overall containment integrity is maintained. Therefore, we limited the public risk calculation to WASH-1400[16] PWR-7 and PWR-9 events, an event similar to a BWR-4 event but with lower containment leakage and less offsite dose consequences, and a BWR-5 event in determining the probability of the events and the potential reduction in offsite consequences. We assumed that the potential offsite dose was 20,000 man-rem/event for the PWR-7 and BWR-4 type of events, 120 man-rem/event for a PWR-9 event, and 20 man-rem/event for a BWR-5 event. These are taken from recent CRAC Code[64] calculations of the integrated whole body man-rem dose to the public in a 50-mile radius of a plant with an average population density of 340 persons per square mile. We assumed a total population of 143 plants (i.e., those now operating plus those yet to be licensed) with an average remaining operating life of 28 years. These assumptions resulted in an estimated probable total dose to the public of slightly over 1,000 man-rem over the lifetime of the reactors.

Cost Estimate

The Working Paper[122] on revising Appendix J and the draft Regulatory Guide,[123] while eliminating the ambiguities in the current regulation, include some proposed provisions that potentially represent more stringent requirements on the licensee. Examples of such provisions are: (1) to require reporting results of unsuccessful containment penetration leak tests as well as successful tests; (2) to require Type A leak tests to be conducted at full pressure rather than at reduced pressure and extrapolated to full pressure; and (3) to provide specific guidance for standardization of acquisition, evaluation, and reporting of leak test data.

We estimated that these changes in leak rate test requirements would result in a net cost to a licensee of an average of $0.5M/yr for additional testing and evaluation of data, reporting, etc. We assumed that the NRC resources necessary to complete revision of Appendix J and issue a Regulatory Guide on leak testing would be balanced by the subsequent reduction of paperwork burden on NRC.

Value/Impact Assessment

Using the information above, the overall value/impact score for PWRs and BWRs is about 0.5 man-rem/$M.

Conclusion

Staff stated in the main report of NUREG-0933 published in 1983 that revising Appendix J and issuing a Regulatory Guide with acceptable containment leakage testing methods had a low potential for reducing risk. However, considering the work accomplished at that time, it was recommended that the containment leakage task be completed as a Regulatory Impact issue on the basis of reducing the compliance burden on licensees and the paperwork burden on the NRC. However, emphasis was placed on eliminating the ambiguities in the present regulation without imposing more stringent leakage testing requirements since they did not appear to be effective in reducing risk.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated

November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

16. WASH-1400 (NUREG-75/014), “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,” U.S. Atomic Energy Commission, October 1975.

64. NUREG/CR-2800, “Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development,” U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

99. ANSI/ANS 56.8, “Containment System Leakage Testing Requirements,” American National Standards Institute, 1981.

122. Working Paper on Appendix J to 10 CFR Part 50, “Leak Tests for Primary and Secondary Containments of Light-Water-Cooled Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, May 17, 1982. [8401040228]

123. Working Paper on Draft Regulatory Guide (MS021-5), “Containment System Leakage Testing,” U.S. Nuclear Regulatory Commission, May 1982. [8405240527]

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

ITEM A-27: RELOAD APPLICATIONS

Description

By letter dated June 18, 1975, licensees of operating reactor facilities were sent a preliminary copy of a staff paper, "Guidance for Proposed License Amendments Relating to Refueling," and "Refueling Information Request Form." The purpose was to provide guidance, although preliminary, to licensees on what information the staff considers to be essential for the conduct of its review of core reload submittals. In order to add more predictability to the review process and to improve the staff scheduling of such reviews, licensees were asked to submit the Refueling Information Request data within 30 days after receipt of the letter and were requested to update the information annually thereafter (or more often if appropriate).

The purpose of this NUREG-0371[2] task is to: (1) update the preliminary guidance issued to licensees in the June 18, 1975 letter to assure conformance with the latest staff technical positions that relate to core reloads, and (2) prepare formal review procedures to assure prompt and uniform review of the licensee reload submittals. Revision of procedures for review of reloads is an important task in order to assure that projected staffing levels will be sufficient to accommodate future reload reviews. Under the present system of individualized reload reviews, the staff level for reload reviews would have to grow proportional to the number of facilities being licensed.

With regard to updating our guidance to licensees, providing licensees with uniform and up-to-date information on our criteria will help to make the review process more orderly and predictable. Ultimately, standardizing the review process will encourage licensees to plan reloads which do not require prior NRC approval and thus will serve to reduce our staffing commitment to reload reviews. Once uniform criteria in the form of the BTP have been developed for use with operating reactors, then a reexamination of the OL stage of licensing will be made to determine if any incentives to licensees exist which would encourage evaluation of reloads prior to receipt of the OL. This would have the effect of allowing the licensee to perform reloads according to certain specifications without NRC approval beyond granting of the OL. In addition, the revised guidance would further underscore our interest in early identification of non-reload related activities which often take place during refueling outages and which require Commission review.

Conclusion

This Licensing Issue was intended to provide a comprehensive guidance document for use by technical reviewers in considering applications for core reloads. A draft Regulatory Guide was issued for comment in September 1979 but was never issued in final form. An NRR Task Force to Address Licensing Reload Reviews reported on November 19, 1981, and recommended preparation of a new Regulatory Guide.[412]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

412. Memorandum for T. Speis from J. Funches, “Prioritization of Generic Issues—Environmental and Licensing Improvements,” February 24, 1983. [8303090540]

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

ITEM B-11: SUBCOMPARTMENT STANDARD PROBLEMS

Description

The calculations of differential pressure that occur in containment subcompartments from a loss-of-coolant event require a complex fluid dynamic analysis to assure that the subcompartment design pressures are not exceeded. To check the various industry computer codes used for the analyses, a standard problem was issued to the reactor vendors and architect/engineers (A/E) so that their models and calculational methods could be evaluated. This NUREG-0471[3] task involved the review and evaluation of the subcompartment standard problem analyses supplied by vendors and A/Es to determine the validity of their models. Changes in the computer codes utilized by the NRC staff could also result from this task.

Conclusion

This item is a Licensing Issue. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM B-13: MARVIKEN TEST DATA EVALUATION

Description

Test data from the Marviken containment tests were obtained for the purpose of validating containment pressure codes used for performing independent calculations related to licensing reviews. The Marviken data are containment pressure responses from a full-scale blowdown using a pressure suppression type containment. This NUREG-0471[3] task would correlate the Marviken data and compare the results with computer programs existed when the issue was identified.

Conclusion

This item is a Licensing Issue. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

Reference

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM B-20: STANDARD PROBLEM ANALYSIS

Description

Most vendors, in the conduct of internal audits of emergency core cooling performance computer codes, discovered errors in coding and/or logic which had significant effects on the prediction results of approved models. This NUREG-0471[3] task involved the use of standard problems to evaluate the predictive accuracy of these complex computer codes and to detect errors, to the extent that the errors affect the results of code predictions.

Conclusion

This item is a Licensing Issue. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM B-25: PIPING BENCHMARK PROBLEMS

Description

Applicants are required to provide confirmation of the adequacy of computer programs used in the structural analysis and design of piping systems and components. At the time this issue was identified, this consisted of applicants providing, and the staff reviewing, brief descriptions of the computer programs used and solutions to simple textbook problems. In order to better provide assurance of the reliability of these programs, this NUREG-0471[3] task would consist of the staff developing benchmark problems and solutions to these problems for use in the review of applications for construction permits. The case-by-case review would then consist of requesting that the applicant submit solutions to the problems and comparing the applicant-supplied solutions to the staff solutions.

Conclusion

As reported in the main report of NUREG-0933 published in 1983, work on this Licensing Issue was being done by BNL.[412] As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

412. Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues—Environmental and Licensing Improvements," February 24, 1983. [8303090540]

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM B-27: IMPLEMENTATION AND USE OF SUBSECTION NF

Description

Since the adoption by the ASME Code, Section III, of Subsection NF on component supports, technical review has been limited to conformance of the information provided in the application and commitment by the applicants to component support design in accordance with the provisions in Subsection NF.

Certain deficiencies in the use of Subsection NF, however, have been identified primarily by NRC Code Committee members on the Working Group on Component Supports and its Task Forces. Examples of these deficiencies are:

(1) The absence of definitive criteria to be used in defining the jurisdictional boundary between a load carrying building structure designed by AISC rules which do not contain inservice inspection requirements and an attached NF component support having NF inservice inspection requirements.

(2) As the design limits for Class 1 liner type component supports presently appear in the Code, the allowable stresses exceed those permitted for other Code-designed components. If these limits are approached repeatedly in the component support, the support could fail by fatigue.

When this issue was added to NUREG-0933 in 1983, it was anticipated that some of the identified deficiencies would be addressed and corrected by revisions to the Code. This NUREG-0471[3] task would develop a BTP that would assess the remaining deficiencies for use by the staff in case reviews of component supports.

Conclusion

This item is a Licensing Issue. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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3. NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D),” U.S. Nuclear Regulatory Commission, June 1978.

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

ITEM B-30: DESIGN BASIS FLOODS AND PROBABILITY

Description

This NUREG-0471[3] task involved the preparation of a paper detailing the bases for design basis flood events utilized by the staff in case reviews, including probable maximum floods, hurricanes, tsunamis, seiches, seismically-induced dam failures, and combinations of lesser events. Additionally, descriptions of probability estimates, including potential errors, would be prepared for the principal flood-producing events. This material was being prepared to respond to a request of the ACRS to provide them with a better understanding of the staff's approach to design basis floods.

Conclusion

This item is a Licensing Issue. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM B-35: CONFIRMATION OF APPENDIX I MODELS FOR CALCULATIONS OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT-WATER-COOLED POWER REACTORS

Description

This NUREG-0471[3] task involves evaluating information from semiannual operating reports, inplant measurements program and topical reports, and revisions to models for calculating releases of radioactive materials in effluents from PWRs and BWRs. This task is expected to improve the accuracy and realism of current staff models by using the best available data to develop model revisions.

Conclusion

As reported in the main report of NUREG-0933 published in 1983, all work on this Licensing Issue was completed except the source term measurement program.[412] As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

412. Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues—Environmental and Licensing Improvements," February 24, 1983. [8303090540]

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM B-49: INSERVICE INSPECTION CRITERIA AND CORROSION PREVENTION CRITERIA FOR CONTAINMENTS

Description

GDC-53, "Provisions for Containment Testing and Inspection," requires in part that the reactor containment be designed to permit: (1) periodic inspection of all important areas, and (2) an appropriate surveillance program. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," requires a general inspection of the surfaces of the containment prior to any Type A test to uncover any evidence of structural deterioration.

Containment designs typically utilize any one of the following structural materials: steel, steel-lined reinforced concrete, and steel-lined prestressed concrete. At the time this issue was added to NUREG-0933 in 1983, the only detailed criteria that were developed for ISI of containments related to tendon surveillance for prestressed concrete containments. These criteria were contained in Regulatory Guides 1.35[481] and 1.90[482] which addressed ungrouted and grouted tendons, respectively. These Regulatory Guides dealt primarily with the prestressing hardware; no detailed ISI criteria existed for the steel liner or other portions of the containment. Similarly, there were no criteria for ISI of steel containments or steel-lined reinforced concrete containments. In view of this, detailed and comprehensive criteria needed to be developed for performing ISI of all types of containments.

In addition, the long-term corrosion problems of reinforcements and of the steel liner in contact with concrete in concrete containments, or the corrosion of the steel surface in contact with the water in BWR suppression chambers, had yet to be adequately analyzed. Long-term studies of these corrosion phenomena needed to be undertaken to develop criteria and requirements to prevent corrosion in all types of containments. This issue is documented in NUREG-0471.[3]

Conclusion

This item is a Licensing Issue. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

3. NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D),” U.S. Nuclear Regulatory Commission, June 1978.

481. Regulatory Guide 1.35, “Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments,” U.S. Nuclear Regulatory Commission, February 1973, (Rev. 1) June 1974, (Rev. 2) January 1976 [7907100149], (Rev. 3) July 1990 [7809180004].

482. Regulatory Guide 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," U.S. Nuclear Regulatory Commission, August 1977. [7907100329]

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM B-72: HEALTH EFFECTS AND LIFE-SHORTENING FROM URANIUM AND COAL FUEL CYCLES

Description

Practice in health impact assessments at the time of identification of this issue was to convert radiation exposure estimates into estimates of health effects, such as cancer deaths, illness, and life shortening. However, the models that were being used, such as those in WASH-1400, GESMO, NRC case-related testimony, and EPA assessments, all suffered from similar weaknesses. A major common weakness, which appeared amenable to solution, was related to the correct treatment of competing risks among populations with life expectancies, age, and sex distributions that vary with time. Since the staff was attempting to assess health effects in the future (e.g., year 2000 and beyond), it was reasonable to expect significant changes in current population statistics. To make such an assessment, a demographic model was required which extrapolated population into the future, correctly allowing for competing risks of mortality from various causes (e.g., accidents, heart disease, and cancer). Failure to do so results, for example, in hypothetical cancer deaths for people who would statistically die from other causes. In the absence of better predictive models, it was not possible to even evaluate the uncertainty associated with the use of the current simplified methods for estimating health effects and consequent life-shortening. Uncertainties in the use of current models were greatly magnified when attempting to make comparisons of health effects for the coal and nuclear fuel cycles.

At the time this issue was added to NUREG-0933 in 1983, health effects models generally were used for estimating long-term impacts. Chronic exposure may be the primary determinant of the number of deaths for a given period for a given pollutant. However, in the case of nonradiological pollutants from the coal fuel cycle, short-term fluctuations leading to acute exposures may determine the time of death and consequent life shortening. Evaluations of the coal fuel cycle generally failed to account for short-term mortality, disease, and illness. In addition, short-term effects from chemical pollutants were generally dependent on the prior history of chronic (long-term) exposure.

Models generally assumed linear dose-response relationships even when evidence existed for real or practical thresholds, or where experimental data supported a nonlinear dose response relationship.

This NUREG-0471[3] task involved the development of models to address these problems so that health effects (morbidity and mortality) could be assessed for both the coal and uranium fuel cycles as completely as data permitted and on a comparable basis.

Conclusion

NRC staff stated in the main report of NUREG-0933 published in 1983 that the results of Item A-20 would be used in this Licensing Issue.[412] As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

412. Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues—Environmental and Licensing Improvements," February 24, 1983. [8303090540]

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ITEM C-14: STORM SURGE MODEL FOR COASTAL SITES

Description

Licensees are required to estimate the design-basis flood levels for each nuclear power plant site consistent with the requirements in General Design Criterion 2, “Design Bases for Protection against Natural Phenomena,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” and 10 CFR 100.20(c). For coastal and estuarine sites, the design-basis flood could be caused by a storm surge that results from the wind and pressure fields of an intense storm. The primary tool used for estimating storm surge at the time when this generic safety issue was identified was the “bathystrophic” model developed by the U.S. Army Corps of Engineers (USACE), Coastal Engineering Research Center. This model, called SURGE, is based on the bathystrophic approximation, relating sea surface slope to wind stress, bottom stress, and pressure gradient, with a correction for Coriolis force on along-shore currents. The SURGE model served its intended purposes well at that time but is now considered obsolete.

The advent of more powerful computing power allows relative quick solutions to numerical the multidimensional hydrodynamic equations which can account for most conceivable physical wave effects that were not included in the SURGE model. In addition, the modern multidimensional hydrodynamic models can account for irregular shorelines and true long-wave hydrodynamics that are not accounted for by the bathystrophic model.

This task under NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D), issued June 1978,[3] called for the development of a replacement for the bathystrophic model so that the staff’s evaluation of storm surge will reflect state-of-the-art techniques. Because the storm surge modeling is applied at the early site permit or combined license stages, only future license applications for nuclear power plants located at coastal or estuarine sites will be impacted by the issue.

Conclusion

This generic safety issue involved the development or application of a multidimensional model that uses state-of-the-art mathematical techniques in estimating hurricane storm surge. The U.S. Nuclear Regulatory Commission staff anticipated that a new multidimensional hydrodynamic model would eliminate the need for initial estimates (required by the bathystrophic model) and would reduce the total required analysis time. Thus, this item is related to increasing knowledge that would increase confidence in the safety assessment; therefore, it is considered to be a licensing issue.

The staff completed a review of this issue in 1988 and concluded that the bathystrophic model (SURGE) existing at the time was adequate for calculating design-basis water levels at proposed nuclear plant sites. The use of SURGE was specified in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (the SRP),[11] Section 2.4.5, Revision 2. However, as stated in the SRP, the use of other verified models was not precluded. Thus, the staff concluded that this licensing issue did not require any changes and recommended dropping the issue from further consideration.

Because of subsequent advances in technology, the staff reexamined the issue in 2010. The staff determined that the primary tools used for estimating storm surge could be the ADCIRC (Advanced Circulation) storm surge model as developed by USACE, the Sea, Lake, and Overland Surges from Hurricanes (SLOSH) model developed by the National Oceanic and Atmospheric Administration (NOAA), or other compatible hydrodynamic models that could simulate both storm surge wind setup and wave runup accurately. These models are based on the hydrodynamic processes of offshore and overland surge wave propagation, relating sea surface slope to wind stress, bottom stress, and pressure gradient, with a correction for Coriolis force on along-shore currents. Powerful computers are capable of producing accurate solutions to multidimensional hydrodynamic equations, which account for many meteorological and hydrodynamic effects of storm surges. True long-wave dynamics are also simulated by multidimensional dynamic mathematical models.

The staff believes that existing hydrodynamic models (e.g., ADCIRC, SLOSH) are adequate for calculating design-basis flood levels at proposed nuclear plant sites. The ADCIRC model has been widely used by the Federal Emergency Management Agency in performing storm surge frequency analyses and by USACE for the design of storm damage reduction projects, while the SLOSH model has been extensively used by the NOAA in predicting storm surge levels for hurricane emergency preparations. SRP Section 2.4.5 and Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," [1981] provide specific guidance in reviewing the applicant's storm surge flood estimates required for new reactor licensing. Therefore, the staff has determined that this conclusion is consistent with the state of the current regulatory framework and technology and does not change the status of this issue.

References

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

11. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," U.S. Nuclear Regulatory Commission, (1st Edition) November 1975, (2nd Edition) March 1980, (3rd Edition) July 1981.

1981. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, June 2007.

ISSUE 59: TECHNICAL SPECIFICATION REQUIREMENTS FOR PLANT SHUTDOWN WHEN EQUIPMENT FOR SAFE SHUTDOWN IS DEGRADED OR INOPERABLE

Description

Historical Background

As a result of the loss of high head injection capability at McGuire Unit 1 on February 12, 1982, this issue was raised by Region II because plant TS require (somewhat rapid) plant shutdown if certain safety equipment is inoperable.[553] The main concern is that the TS requirements may not adequately consider the potential for placing the plant in a "less safe" condition by requiring shut-down of an otherwise normally functioning unit or by requiring a plant to proceed to cold shutdown when "hot shutdown" may be the more desirable condition.[767]

Safety Significance

Plant TS LCOs are typically written to require proceeding to various stages of shutdown if certain systems are inoperable. If some systems are inoperable and a plant is required by the TS to go to some stage of shutdown, this action may increase the probability of needing the inoperable systems as a line of defense. In some cases, the shutdown process itself may require operation of the inoperable equipment.

Possible Solution

A resolution could require TS modifications to acknowledge when continued power operation or other mode of operation is preferable. Because of the wide range of possible system failures, operating conditions, and plant configurations, a systematic quantification of all the alternatives could be a fairly large task and would probably result in a number of decisions based on very close calls.

Priority Determination

Assumptions

In order to provide some indication of what a quantitative analysis may involve, an assessment of this issue was done by PNL.[64] A number of assumptions were made in this analysis in an attempt to quantify a potential core-melt frequency reduction for a case in which a plant was left operating, as opposed to rapidly shutting it down, given some safety equipment is inoperable.

The ORNL Precursor Study (NUREG/CR-2497)[76] was used along with data from an EPRI ATWS study (NP-2230)[307] to calculate a base case and and adjusted-case core-melt frequency for this issue. The techniques and data presented in NUREG/CR-2497[76] were modified to allow a comparison of the risk of core damage with a safety system inoperable for continued reactor operation versus immediate shutdown. To accomplish this, specific systems were chosen for failure and appropriate event trees developed. Data on system failure were then adapted to fit the need for: (1) failure frequency, (2) failure on demand, or (3) failure over a specified time interval.

For this analysis, a BWR was chosen and it was assumed that the HPCI and RCIC are redundant safety systems. Failure of both the RCIC and HPCI was then postulated.

To examine this issue, generic event trees were developed based on the flow logic developed in the Precursor Study[76] for BWR transients. The first event tree[64] depicted a failure of safety systems followed by a shutdown by the operator. The transient which could then follow was shown as "loss of feedwater given shutdown," chosen here as representative of transients which would challenge the ECCS. The second event tree depicted the case where operation continues. Another initiating event is then required, taken here as loss of feedwater given an ECCS subsystem failure. The following data were taken from NUREG/CR-2497:[76]

Event Description	**Occurrences**	**Plant-Years (RY)**	**Demands**	**Failure Frequency$(RY)^{-1}$**	**Failure Probability on Demand**
Loss of Feedwater	39	66	-	0.58	-
Reactor Subcritical	-	-	-	-	1.3×10^{-6}
RCIC/HPCI Failure	4.9	99	-	0.049	0.0039
Long-Term Core Cooling Failure	-	-	-	-	1.1×10^{-4}

The analysis results hinge on the probability of inducing a feedwater transient on shutdown vs. a feedwater transient occurring at power during the time the systems remain inoperable. Data for these values are lacking at this time so values are estimated based on the ATWS report.[307] For BWR Transient Category 26 (decreasing feedwater flow during startup or shutdown), the frequency reported is 0.07/RY. It is assumed here that the plant is shutdown about 12 times per year resulting in a probability of about 0.01 for a feedwater transient on shutdown. It is further assumed that 50% of these transients are decreases in feedwater during shutdown with 50% of these resulting in complete loss of function. The probability (p_1) of loss of feedwater on scram is therefore assumed to be (0.01)(0.5)(0.5) = 0.0025.

To estimate the probability of feedwater failure during an ECCS subsystem outage, a one-day failure duration is assumed with the plant remaining at power for that 1 day. The probability (p_2) of independent loss of feedwater over the one-day ECCS subsystem outage is approximated by $p_2 = \lambda t = 0.0016$, where $\lambda = 0.58$/RY and t = (1 day)/(365 day/RY) = 0.0027 RY.

These data were entered in the event trees by PNL,[64] Sequences 5, 6, 7, and 8 were summed and then Sequences 12, 13, 14, and 15 were summed. The core-melt frequency was calculated to be 3.5×10^{-6}/RY for the base case, i.e., the rapid shutdown of the plant. The core-melt frequency was calculated to be 2.2×10^{-6} /RY for the adjusted case, i.e., where the plant continued to operate.

The event trees developed for the Precursor Study[76] give a measure of core damage only. To equate this with the core-melt frequency used in other risk studies, the above core-damage frequencies were divided by a factor of 30 for the reasons given below.

An analysis of the ORNL Precursor Study by INPO claims that the chances of a severe nuclear accident were estimated 30 times too high.[64] Furthermore, severe core damage (assumed to be analogous to that at TMI-2 in the Precursor Study) is presumably less severe than core-melt, the level of core damage normally considered in nuclear power plant risk studies. Based on these considerations, it is assumed that the frequency of core damage as assessed using the Precursor Study should be divided by INPO's factor of 30 to result in the frequency of core-melt.

Thus, the base case and adjusted case core-melt frequencies become 1.2×10^{-7}/RY and 7×10^{-8}/RY, respectively and the core-melt frequency reduction is 5×10^{-8} /RY. An average LWR dose factor of 3.3 x 10* man-rem was calculated from NUREG/CR-2800,[64] (Appendices A-D). Based on this factor, the potential risk reduction would be (5×10^{-8}/RY)(3.3×10^{6} man-rem) or 0.17 man rem/RY.

The result shows a slight decrease in risk. However, the calculation is heavily dependent on the assumed value for the probability of loss of feedwater on shutdown vs. the probability of loss of feedwater over a 1 day ECCS outage. This dependency can be seen by doing the same type of calculation but assuming a 1.5 day outage time. Then, for the adjusted case, $P = \lambda t = 0.0024$ and evaluating events 12, 13, 14, and 15 yields an adjusted case frequency of about 3.2×10^{-6} /RY. This would then show an even smaller risk reduction when compared to the base case result of 3.5×10^{-6}/RY. (Again, these would be reduced by a factor of 30). Similarly, if 2 days were assumed, $P = \lambda t = 0.32$ and the core-melt frequency would be about 4.3×10^{-6}/RY which would show a slight increase in risk for staying at power. These calculations show the sensitivity of the results to the assumptions and the data.

Cost Estimate

Industry Cost: The direct cost would be $4,000/plant for a Class III amendment to an operating license. Other costs for implementation could be significant for analysis of various plant situations to identify the preferred mode and, therefore, justify the change.

Based on 71 operating plants, the industry cost was assumed to be (71 plants) ($4,000/plant) or $0.28M and 1 man-year/plant for supporting analysis or ($100,000/plant)(71 plants) = $7.1M. Since most changes would involve a justification for continued power operation, a potential large cost saving could be involved. For calculation purposes, it could be assumed that over the life of a typical plant, at least 1 day of shutdown may be saved. At $300,000/ day, the industry cost saving for 134 plants is $40M.

NRC Cost: NRC cost for issue development was based on the assumption that considerable analysis (and review of licensee submittals) would be needed to quantify safety benefits associated with TS modifications. This was assumed to be about 3.5 man-years or $3.5M.

Other Considerations

1. Since this issue was originally raised, the NRC has published a rule which allows relief from TS requirements in an emergency situation. This rule leaves the decision to the licensees of determining: (1) what constitutes an emergency, and (2) what is the most prudent action to take. During the comment period on the rule, it was requested that comments be provided regarding whether or not the rule should have more specific guidance. It was concluded, based on comments received, that it was not feasible to provide detailed guidance as to when deviations are permissible. It was felt that this would

defeat the purpose of the rule which is to provide flexibility in situations that cannot always be anticipated.

2. More recently, the general issue of whether TS are properly focused or are unduly burdensome has been raised. In response to this problem, a Technical Specification Improvement Project has been established.[768] This project will consider the safety relevance and burden of the TS as a whole and of specific sections. This issue is one example of a possible modification to improve the TS.

3. The above analysis was done based on assuming a situation in which a plant is at power and the question is whether to require the plant to proceed to shutdown. It was pointed out that a clearer case could be made for situations of the plant being in hot shutdown and requiring proceeding to cold shutdown. Regardless, both situations could lead to potentially large cost savings for the industry and it may be to a licensee's advantage to try to anticipate the possibility of these situations and submit modified TS to avoid crisis-type decisions (i.e., emergency TS relief) when the emergency arises or to avoid second-guessing after the emergency passes if, for example, the rule is used.

4. The McGuire event (which is part of the basis for raising this issue) could have been a case for application of the rule. The question would have been if, as postulated, the situation would have continued (i.e., no charging/SI pumps), would it have been preferable for the licensee to deviate from the TS and keep the plant on line and, if so, how long should power operation be continued?

5. The situations of concern are typically beyond the design bases of the plant and, therefore, should occur rather infrequently.

6. For cases like McGuire where shutdown would require the inoperable equipment, it appears that a TS change may not solve the problem because, no matter what length of time is chosen for continued operation, there is some probability that the equipment would not be restored in the allowed time and shutdown would be necessary anyway (either due to the TS or a transient). For such cases, it is probably best to let the licensee use the rule based on a consideration of the specific plant circumstances at that time. It should be pointed out that the AFW system TS 3.7.1.2 (which was suggested as a solution[553]) was written prior to the Rule[767] and would probably not be included in the TS if the rule had been in effect at that time.

Conclusion

Although we did not calculate a specific value/impact score for this issue, the calculation of potential man-rem reduction for the assumed scenario gave an indication of the uncertain nature of this type of analysis. After consideration of the new rule, we concluded that to a large extent the safety implications of this issue have been addressed. The rule gave the licensees the flexibility to consider their individual plant circumstances and make a decision to deviate from the TS if they decide it is necessary. However, as has been pointed out, there may be specific cases where changes should be considered.[767] Because the risk was so hard to quantify, we originally assumed a small change in public risk, acknowledged the potential cost saving, and concluded it should be a Regulatory Impact issue to be addressed by the Technical Specification Improvement Project.[768]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

76. NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report," U.S. Nuclear Regulatory Commission, June 1982.

307. EPRI NP-2230, "ATWS: A Reappraisal, Part 3," Electric Power Research Institute, 1982.

553. Memorandum for D. Eisenhut from J. Olshinski, "Loss of High Head Injection Capability at McGuire Unit 1 and Reconsideration of Technical Specification 3.0.3 and 3.5.2," April 12, 1982. [8802120046]

767. Memorandum for W. Dircks from H. Denton, "Final Rule—Applicability of License Conditions and Technical Specifications in an Emergency," February 17, 1983. [8303300333]

768. Memorandum for T. Speis from H. Denton, "Formation of a Technical Specification Improvement Project Group," December 31, 1984. [8501150417]

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ISSUE 67: STEAM GENERATOR STAFF ACTIONS

Description

Following the SGTR event at Ginna on January 25, 1982, increased staff effort was placed on developing means to mitigate and reduce steam generator tube degradations and ruptures. To meet these objectives, two steps were taken. The first step was to develop staff requirements to be implemented by licensees; these were evaluated in Issue 66. The second step was to develop recommendations for staff action; these were evaluated below.

The status of these actions as determined in this evaluation is listed in Table 3.67-1. For reference proposes, the sub-item numbers are consistent with the staff action numbers provided by DL/NRR.[752] These items are also included in the CRGR review package[753] and EDO recommendations to the Commission.[753,757,758] The following is a summary of the evaluation of the 16 parts of this issue.

(a) Three of the proposed staff actions were classified as Licensing Issues:

5.1 Reassessment of Radiological Consequences
5.2 Reevaluation of SGTR Design Basis
10.0 Supplemental Tube Inspections

(b) Two of the proposed staff actions were classified as Regulatory Impact issues that could provide cost benefits to the NRC and industry:

2.1 Integrity of Steam Generator Tube Sleeves
8.0 Denting Criteria

(c) Nine of the proposed staff actions were considered part of existing staff activities and needed no new staff efforts to be initiated:

3.1 Steam Generator Overfill
3.2 Pressurized Thermal Shock
3.3 Improved Accident Monitoring
3.4 Reactor Vessel Inventory Measurement
4.1 RCP Trip
4.2 Control Room Design Review
4.3 Emergency Operating Procedures
6.0 Organizational Responses
9.0 Reactor Coolant System Pressure Control

(d) The improved Eddy Current Tests (Item 67.7.0) recommendation was integrated into the resolution of Issue 135. The remaining proposed staff action (Item 67.5.3) was placed in the drop category.

The basis for each of the 16 recommended staff actions is provided in separate evaluations below.

TABLE 3.67-1

Sub-Item	Staff Action	Priority	MPA No.
67.2.1	Integrity of Steam Generator Tube Sleeves	135	NA
67.3.1	Steam Generator Overfill	A-47, I.C.1	NA
67.3.2	Pressurized Thermal Shock	A-49	NA
67.3.3	Improved Accident Monitoring	NOTE 3(a)	A-17
67.3.4	Reactor Vessel Inventory Measurement	II.F.2	F-26
67.4.1	RCP Trip	II.K.3(5)	G-01
67.4.2	Control Room Design Review	I.D.1	F-08
67.4.3	Emergency Operating Procedures	I.C.1	F-05
67.5.1	Reassessment of Radiological Consequences	LI(NOTE 3)	NA
67.5.2	Reevaluation of SGTR Design Basis	LI(67.5.1)	NA
67.5.3	Secondary System Isolation	DROP	NA
67.6.0	Organizational Responses	III.A.3	NA
67.7.0	Improved Eddy Current Tests	135	NA
67.8.0	Denting Criteria	135	NA
67.9.0	Reactor Coolant System Pressure Control	A-45, I.C.1(2,3)	F-04, F-05
67.10.0	Supplemental Tube Inspections	LI(NOTE 5)	NA

ITEM 67.2.1: INTEGRITY OF STEAM GENERATOR TUBE SLEEVES

Description

Historical Background

This item was Recommendation 2.1 of the DL/NRR memorandum[752] and called for the staff to develop an SRP[11] Section to clarify staff positions on the materials design, fabrication, installation, examination, and inspection of steam generator tube sleeves.

Safety Significance

At the time this issue was raised, there was no specific SRP[11] Section to guide the staff/industry in reviews related to the design, installation, and inspection of tube sleeves. Development of an

SRP[11] would provide an acceptable means to meet GDC 14 and GDC 32 of 10 CFR 50, Appendix A.

Priority Determination

Consequence Estimate

The public risk reduction attributable to this recommendation was not quantifiable. It was believed that some small improvement in the effectiveness of sleeves to perform their intended function (i.e., assure retention of structural integrity of degraded tubes) could result from improved guidance.

Cost Estimate

Three man-months of NRC staff time ($25,000) were estimated for the development of the SRP.[11] It was estimated that 25% of the operating and planned PWRs (22 plants) would require tube sleeve modifications. The SRP[11] could reduce plant-specific reviews from 2 man-months to 1 man-month and was expected to also reduce industry manpower requirements by approximately the same amount. Therefore, the SRP[11] would result in cost savings of $158,000 and $183,000 to the NRC and industry, respectively, for a combined saving of $341,000.

Conclusion

A small public risk reduction was achievable from development of an SRP[11] on steam generator tube sleeves. However, the SRP would be cost-effective in that it would reduce NRC review cost and industry costs associated with the design, installation, and inspection requirements for tube sleeves. The earlier the SRP[11] was developed, the greater the cost saving. This issue was addressed in the resolution of Issue 135.[1075]

ITEM 67.3.1: STEAM GENERATOR OVERFILL

Description

Historical Background

This item was Recommendation 3.1 of the DL/NRR memorandum[752] and called for the NRC to select a small number of PWRs representing the PWR spectrum of designs and determine the potential for, and consequences of, steam generator overfill as a result of an SGTR. This recommendation was closely related to Items 67.5.1, 67.5.2, and 67.9. Based on the results of these studies, further NRC or licensee actions were to be determined. Potential steam generator overfill resulting from control system failures were not considered in this recommendation. Steam generator overfill via control systems failures were evaluated in the resolution of Issue A-47; Issues 37 and 56 were also related issues.

Safety Significance

Following an SGTR, the affected steam generator could fill up to the steam line safety valve due to primary-to-secondary leakage from continued operation of the safety injection pumps. The safety valve could lift at successively lower pressures and fail to fully reseat. The failure to completely reseat could contribute to steam generator overfill by lowering the damaged steam generator pressure, thus raising the differential pressure across the broken tube and sustaining the leakage despite reduced primary system pressure. Failure of the valve to reseat would also provide a direct pathway for release of radioactive primary water to the environment. This sequence of events is beyond the design basis for SGTR events in SRP[11] Section 15.6.3 to establish that the radiological consequences meet 10 CFR 100.

For the B&W OTSG design in particular, it may not be possible to stop the primary-to-secondary leakage in an SGTR while maintaining the RCS in a subcooled state. The increased tendency for the OTSG leakage to continue throughout the event is a result of the tubes being directly exposed to the OTSG steam space. Generally, the emergency procedures instruct the operator to discharge steam to the atmosphere or, if available, to the condenser to control level in the damaged steam generator, as necessary. However, in at least one B&W plant, if the water supply for safety injection pumps is approaching a minimum level or if the offsite radiological consequences are becoming excessive, the OTSG is allowed to completely fill, thus terminating the leakage. The number of B&W plants that permit filling of the OTSG was not known. The staff did not believe that the potential for prolonged leakage and the associated offsite radiological consequences had been factored into OR or NTOL FSAR SGTR accident analyses. (See Item 67.5.2).

Possible Solutions

Solutions could involve improved RCS pressure control to reduce the differential pressure and leakage across the broken steam generator tube (primary to secondary), and/or improved EOPs to preclude overfill. The above measures were discussed in response to staff recommendations concerning RCS pressure control and EOPs. (See Items 67.9.1 and 67.4.3). With regard to the concern that the steam lines cannot support the dead-weight load if the lines are filled with water, additional supports or stronger steam lines could resolve this aspect of the concern.

Priority Determination

Cost Estimate

The NRC cost would be dependent on the number of PWRs selected for this study and the design variations within this selected group.

Other Considerations

Following the Ginna event, concerns were raised relative to the potential for failing the steam lines under the additional dead-weight load if they are filled with water as a result of steam generator overfill. (The Point Beach SGTR, which was a relatively low leak rate, resulted in a near overfill condition.)[755] Should the steam lines fail, the SGTR could become a LOCA outside containment. However, analyses[753] conducted for 4 plants indicated that the steam lines were unlikely to fail under the additional dead-weight load. Accordingly, the staff's risk analyses[753] assumed a conditional probability of steam line break, given a steam generator overfill, of 10^{-3} which was believed to be reasonably conservative. If the steam lines were re-designed to withstand an overfill condition, the analysis[753] would indicate a reduction in core-melt frequency of 1.2×10^{-7}/RY.

The consequence resulting from failure of the steam lines by overfilling the steam generators was assumed to involve releases typical of a PWR Category 4 release. Exposure was calculated assuming a typical mid-West meteorology and a population density of 340 persons/square-mile within a 50-mile radius of the plant. The potential public risk reduction was therefore [$(1.2 \times 10^{-7})(2.7 \times 10^{6})$] man-rem/RY or 0.32 man-rem/RY. Considering an average remaining plant life of 24 years, the public risk reduction was about 8 man-rem/reactor.

Conclusion

This item encompassed several considerations related to steam generator overfills and was closely related to staff studies identified in Items 67.5.1, 67.5.2, and 67.9. The primary concern (mitigation of a steam generator overfill) was part of the following existing staff programs: (1) Issue A-47; and (2) NUREG-0737,[98] Item I.C.1 (See Item 67.4.3). Therefore, the steam generator overfill issue was covered by the above programs. Rupture of steam lines as a result of a steam generator overfill is a secondary concern predicated on the condition that an overfill occurs. The public risk associated with rupture of steam lines is low and strengthening of the steam lines was considered a LOW priority.

ITEM 67.3.2: PRESSURIZED THERMAL SHOCK

Description

Historical Background

This item was Recommendation 3.2 of the DL/NRR memorandum[752] and called for the staff to address the effects of RCS flow stagnation associated with isolation of a steam generator in the Pressurized Thermal Shock program (Issue A-49).

Safety Significance

During the Ginna SGTR event, the affected steam generator was isolated and the RCPs were tripped. As a result, the flow in the `B' Reactor Coolant Loop was reduced to a few hundred gpm while cold high pressure injection water was being injected into the loop. The cold leg piping apparently experienced a cool-down of approximately 260°F in 30 minutes. The reactor vessel apparently did not experience this rapid cool-down since the flow in the cold leg was in the reverse direction, i.e., from the reactor vessel towards the steam generator. Other events, as discussed in NUREG-0916,[754] resulting in a steam generator isolation and continued safety injection could result in adding cold water to the reactor vessel.

Conclusion

The probability, consequences, and resolution of the above events were addressed in Issue A-49.

ITEM 67.3.3: IMPROVED ACCIDENT MONITORING

Description

Historical Background

This item was Recommendation 3.3 of the DL/NRR memorandum[752] and called for the staff to address the accident monitoring weaknesses of the type observed at Ginna by implementation of Regulatory Guide 1.97[55] and the SPDS.

Safety Significance

During the event at Ginna, several weaknesses in accident monitoring were apparent. These included: (1) non-redundant monitoring of RCS pressure; (2) failure of the position indication for the steam generator relief and safety valves; and (3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents. These conditions make it more difficult for correct operator action in response to such events.

Possible Solution

Had Regulatory Guide 1.97[55] been implemented at Ginna before the January 1982 event, the monitoring of the event would have been substantially improved and there would have been more assurance of correct operator actions. Improved accident monitoring would also have improved the NRC's ability to assess the plant status and the appropriateness of the licensee's actions and recommendations.

Conclusion

This issue was covered in Supplement 1 to NUREG-0737[98] (Generic letter No. 82-33)[376] and was RESOLVED and implemented as MPA A-17.

ITEM 67.3.4: REACTOR VESSEL INVENTORY MEASUREMENT

Description

Historical Background

This item was Recommendation 3.4 of the DL/NRR memorandum[752] and called for implementation of TMI Action Plan Item II.F.2 because it would have substantially improved the Ginna situation by ensuring that steam bubble formation in the reactor vessel upper head could be more accurately monitored.

Safety Significance

During the Ginna SGTR event, the formation of a steam bubble in the reactor vessel upper head significantly complicated the course of the event. The uncertainty about the bubble size was a significant factor in the operator's decisions to continue safety injection beyond the point when termination was called for in the emergency procedures.

Possible Solution

Implementation of NUREG-0737,[98] Item II.F.2.

Conclusion

Following Commission approval for implementation of Item II.F.2, letters to individual licensees and orders to B&W licensees and ANO-2 were issued on December 10, 1982.[491] Thus, this issue was covered in Item II.F.2 which was resolved and implemented as MPA F-26.

ITEM 67.4.1: REACTOR COOLANT PUMP TRIP

Description

Historical Background

This item was Recommendation 4.1 of the DL/NRR memorandum[752] and called for the NRC to develop requirements for licensees to provide RCP trip criteria that would ensure continued forced RCS flow during steam generator tube breaks, up to and including the design basis tube rupture.

Safety Significance

Analyses indicated that continued operation of the RCPs following a range of small LOCAs could lead to excessive inventory loss for which the high pressure injection system would be unable to compensate. Generally, the range of break size of concern was from 0.02 to 0.2 ft^2 (2 to 5 inches equivalent diameter). The interim position (documented in NUREG-0623)[97] required manual-tripping of the RCPs on the symptoms of a small LOCA (i.e., a safety injection signal and low RCS pressure).

Conclusion

This issue was covered in NUREG-0737,[98] Item II.K.3(5), which was resolved and implemented as MPA G-01.

ITEM 67.4.2: CONTROL ROOM DESIGN REVIEW

Description

This item was Recommendation 4.2 of the DL/NRR memorandum.[752] As a result of a review of the Ginna control room following the tube rupture, several items related to the event were identified that were contrary to good human factors engineering principles. It was recommended that these items be reviewed by HFEB/NRR as part of the detailed control room design review required by NUREG-0737[98] and the information used as the basis for a study to determine what changes could be made to improve control room designs.

Conclusion

This issue was covered in NUREG-0737,[98] Item I.D.1, which was resolved and implemented as MPA F-08.

ITEM 67.4.3: EMERGENCY OPERATING PROCEDURES

Description

Historical Background

This item was Recommendation 4.3 of the DL/NRR memorandum,[752] the purpose of which was to ensure that newly-developed EOPs consider the experiences from the Ginna SGTR event. PSRB/NRR was expected to review the items listed below prior to emergency procedure implementation for inclusion in its review plan. This staff effort was to be considered in conjunction with existing work on NUREG-0737,[98] Item I.C.1.

- RCP Restart
- Availability of Faulted Steam Generator Safety and Relief Valve
- Multiple and Second Order Failures
- Bubble Formation
- Cooling Faulted Steam Generator
- Cooling Intact Steam Generator
- Safety Injection Pump Termination and Restart Criteria
- Procedure Format and Clutter
- Criteria for Natural Circulation Determination
- Accommodation of Plant Differences from Reference Plant in Emergency Procedure Development
- Rapid Determination and Isolation of Faulted Steam Generator and Timely Depressurization of RCS to Minimize RCS Inventory Loss and Releases
- MSIV Closure During Plant Cooldown
- Use of Charging and Letdown Systems
- Operation of the RCP in the Damaged Loop
- Operation of Loop Isolation Valves
- Use of Pressurizer PORV
- Potential Complicating Events
- Site-Specific Operator Training
- Steam Generator Level Control for CE Plants

Safety Significance

The above list included transients and plant conditions that form the basis of many of the emergency procedures, reliability analyses, human factors engineering, crisis management, and operator training. Plant conditions may exist, in addition to those pertinent to design bases, which could prevent proper operator actions during such events/conditions and possibly pose a serious threat to reactor safety.

Possible Solution

The solution to this recommendation was to consider the Ginna event in the development of EOPs.

Priority Determination

Guidance for the evaluation and development of procedures for transients and accidents was covered by Item I.C.1 of NUREG-0737.[98] Some of the items in the above list were explicitly included in the review requirements of Item I.C.1. Other items in the list are believed to be implicitly within the intent of Item I.C.1 in that the availability of systems under expected conditions (like Ginna) should be used in developing diagnostic guidance for operator and procedural development.

Conclusion

This issue was covered in NUREG-0737,[98] Item I.C.1, which was resolved and implemented as MPA F-05.

ITEM 67.5.1: REASSESSMENT OF RADIOLOGICAL CONSEQUENCES

Description

Historical Background

This item was Recommendation 5.1 of the DL/NRR memorandum[752] and called for the staff to reassess SGTR events at W and CE plants only to determine the effects of releases made for periods substantially longer and via other release points than those previously analyzed. These analyses should specifically address the applicability of the assumptions in SRP[11] Section 15.6.3 and address the costs and benefits of requiring revised analyses by licensees. This issue was closely related to Items 67.5.2 and 67.3.1.

Safety Significance

Public risk from an SGTR, even considering steam generator overfill, was considered low for a typical PWR. This low risk was expected to remain valid even if new source term results were applied. However, the safety significance of this issue was derived from concern over the number of SGTR events and potential for exceeding the bounds of the analyses that are currently required in SRP[11] Section 15.6.3 to demonstrate that doses from SGTR events will not exceed 10 CFR 100.

Priority Determination

SRP[11] Section 15.6.3 does not address a steam generator overfill in the SGTR scenario. In addition, termination of the leak from an SGTR within 30 minutes, as assumed in typical PWR FSARs, may be non-conservative and not consistent with operating experience. Therefore, implementation of this recommendation would allow the staff to upgrade SRP[11] Section 15.6.3 and provide a better understanding and means to assess future SGTR events in operating plants relative to the consequence limits in 10 CFR 100.

Information generated from implementation of this recommendation would also assist licensees in their understanding of similar events and help determine the course of action needed to mitigate the consequences of SGTRs and overfilling of the steam generators.

Conclusion

Resolution of this issue was not expected to result in significant public risk reduction and, therefore, it was considered a low priority. However, AEB/NRR considered it a Licensing Issue and recommended the reassessment. DST/NRR agreed that a "best estimate" analysis modeled after plant experience like Ginna could be beneficial in more realistically determining the risk and conservatisms inherent in the existing SRP[11] requirements. The issue was finally resolved[1554] with recommended changes to SRP[11] Section 15.6.3.

ITEM 67.5.2: REEVALUATION OF SGTR DESIGN BASIS

Description

Historical Background

This item was Recommendation 5.2 of the DL/NRR memorandum[752] and called for the NRC to reevaluate and consider reclassifying or redefining the design basis SGTR event. This issue was closely related to Issues 67.3.1 and 67.5.1.

An SGTR accident is one of the events for which the NRC requires a safety analysis to show that a reactor will respond in an acceptable manner and that the health and safety of the public are adequately protected. The SGTR accident is the loss of integrity (development of a leak) in a steam generator tube (or tubes) so that reactor coolant water from the primary system flows into the secondary water in the steam generator. This provides a potential path for the release of radioactivity to the environment.

As analyzed in SARs, the event is a break of a single steam generator tube with flow out of the full-flow area of both ends of the steam generator tube at the break. The reactor is assumed to be at full power at the time of the accident.

The SGTR accident serves as the design basis for allowable reactor coolant activity since the amount of radioactivity released to the environment is directly proportional to the amount of activity in the coolant. The analysis of this event in SARs is intended to bound the potential release of radioactivity, should an SGTR occur. The behavior of reactor systems during this event has not traditionally received much emphasis, either in the analyses reported by the licensees or during review by the NRC.

Safety Significance

The safety significance of this recommendation was derived from the concern over the number of SGTR events and the potential for exceeding the bounds of the analyses that were required in SRP[11] Section 15.6.3 to demonstrate that doses from SGTR events will not exceed 10 CFR 100.

Priority Determination

The analysis of an SGTR is performed to bound potential offsite doses using many conservative assumptions (i.e., accident terminated within 30 minutes) to maximize the predicted doses (SRP[11] Section 15.6.3). The probability of the simultaneous occurrence of the SRP[11] conditions is extremely low. SGTR events have occurred at a frequency of approximately 2×10^{-2}/RY. Therefore, this event could be classified as an incident that might occur during the lifetime of a particular plant.

SGTR events that have actually occurred were not as severe as the SRP[11] design basis event. Had the frequencies of the conservative assumptions been included in a calculation of a design basis frequency, a much lower frequency would result. A change in classification would necessarily require changes to the conservative analysis assumptions (listed in the SRP[11]). Changes to the design basis assumptions may include more conservative limits on the reactor coolant activity for those plants that do not have STS limits on coolant iodine concentrations, SGTR overfill conditions, multiple ruptures of the steam generator tubes, and other conditional failure scenarios.

Conclusion

The basis for this issue was derived from the number of SGTR events that occurred and the existing potential for doses from these events that exceeded 10 CFR 100 guidelines. However, these doses would occur only if there were an unlikely (but not impossible) set of circumstances as discussed in detail in Section 8.1 of NUREG-0916.[754]

For the 4 SGTRs that occurred, there were no significant consequences to the public and the existing design basis SGTR was proven to be adequate. The staff believed that it was premature to establish a priority for reclassification of the design basis SGTR event, prior to obtaining the results from other Staff Actions (See Item 67.5.1). Therefore, this issue was considered a Licensing Issue and was integrated into the resolution of Item 67.5.1.

ITEM 67.5.3: SECONDARY SYSTEM ISOLATION

Description

Historical Background

This item was Recommendation 5.3 of the DL/NRR memorandum[752] and called for the NRC to reevaluate the provisions for isolating the steam generators in conjunction with Items 67.3.1 and 67.5.1. The evaluation was expected to consider whether the existing provisions for isolating the main steam and feedwater lines were adequate, with particular emphasis on isolation of the steam generator with RCS loop isolation valves that utilized closed bonnet secondary safety valves or contained the discharge from the steam generator safety and relief (atmospheric dump) valves.

Safety Significance

The primary safety significance of SGTR events is the potential for a direct path for a loss of radioactive coolant from the RCS through the steam generator to outside the containment. This event could also increase the probability of a core-melt because the reactor coolant leaking from a steam generator tube cannot be recirculated. Other systems that penetrate the containment and interface either with the RCS or the containment have two containment isolation valves that close automatically or are locked closed. The steam generator safety and atmospheric valves open automatically and, as required by the ASME Code, cannot be isolated.

Possible Solution

Some of the older PWRs have block valves in the reactor coolant loops that could be used to isolate the steam generators and prevent the loss of coolant and radioactivity from the RCS. Alternatively, the discharge from the steam generator safety and relief valves could be routed to return to the containment or a quench tank. GDC 57 requires each line that penetrates containment (and is neither part of the RCS nor connected to the containment atmosphere) to have at least one isolation valve that is locked closed, automatic, or capable of remote operation. GDC 57 was not interpreted to apply to the valves on the steam generator. However, some improved means of isolating the steam generator, possibly either by requiring loop isolation valves in the RCS or containment of the safety valve discharge, could be considered.

Priority Determination

Recommendation 8 of NUREG-0651[755] stated "... For those plants provided with loop isolation valves, the use of these valves following an SGTR should be investigated. Isolating the affected loop would provide an almost immediate abatement of steam generator tube leakage, but would prohibit cool-down of the damaged steam generator. Licensees should, therefore, examine the advantages and disadvantages in their plant of loop isolation." As pointed out in NUREG-0651,[755] the determination and isolation of the damaged steam generator appeared to take longer than the assumed 30 minutes in the FSAR analysis. In this regard, Item 67.5.1 could address this aspect of steam generator isolation.

The EOPs involved with isolation of the secondary system following an SGTR were identified in Item 67.4.3 as selected events for staff review. In isolating the steam generator, an operator's worst error could be isolating the wrong steam generator. If this were to occur, overfill of the broken steam generator could still result. In addition, the intact steam generator which is isolated could boil dry. Saturated conditions in this hot leg could result. When the operator recognizes the error, isolates the faulted steam generator, and opens the intact steam generator, he might have no steam generator cooling since natural circulation might have become inhibited through the intact steam generator due to void formation. The faulted steam generator would then be isolated, resulting in minimal transfer of heat. The operator could unisolate the faulted steam generator and steam either to the condenser (if available) or to the atmosphere, but this would result in increased offsite doses.

The W SGTR guidelines contain a note that advises operators not to use the loop isolation valves in the event of an SGTR. They further state that "... any use of LSIVs (Loop Stop Isolation Valves) must be justified on a plant-specific basis." The reasons given by W for not using these valves were: (1) their use has not been included in any accident analyses; (2) they are not meant to be safety components; (3) their use has not been recommended, since steam generator isolation has not been shown necessary to limit releases to an acceptable value; (4) the valves are very slow acting and take minutes to close; and (5) their subsequent re-opening required a rather careful procedure.

Conclusion

Many PWRs do not have LSIVs for use in an SGTR accident. For those plants that have them, modifications would likely be required. However, based on the above discussion, the valves did not appear to be necessary. In each of the SGTR events that occurred, the operator took correct action and in none of the events did incorrect action result in any significant adverse effect to the public. In each event, the SGTR was isolated to the faulted steam generator. Therefore, this issue was placed in the DROP category.

ITEM 67.6.0: ORGANIZATIONAL RESPONSES

Description

Historical Background

This item was Recommendation 6.0 of the DL/NRR memorandum[752] and called for the staff to establish, as soon as possible, improved NRC emergency preparedness to handle nuclear accidents at licensed reactor facilities.

Safety Significance

In the event of a nuclear accident, improved NRC emergency preparedness procedures will enable NRC to monitor and evaluate the situation and its potential hazards, advise the licensee's operating staff as needed, and, in an extreme case, issue orders governing such operations.

Possible Solution

Resolution of this item centered around implementation of TMI Action Plan Item III.A.3.

Conclusion

This issue was covered in TMI Action Plan Item III.A.3.

ITEM 67.7.0: IMPROVED EDDY CURRENT TESTS

Description

Historical Background

Improved Eddy Current Tests (ECT) were originally proposed by the staff as requirements to be implemented by licensees. Improved ECT could enhance earlier detection of degradations and thereby minimize, or mitigate, steam generator tube degradations and ruptures. The evaluation of improved ECT as a requirement (Item 66.3) showed that use of current state-of-the-art improvements provided only small reductions in public risk. Likewise, since ECT was an evolving technology, imposition of any requirement was determined to be premature. However, it was also recognized that significant potential reductions in ORE could result from use of improved ECT. Therefore, this item was believed to warrant a medium priority ranking. The conclusion reached in

Item 66.3 was consistent with the position that improved ECT should be handled as a Staff Action item and developed in accordance with the possible solution described below.

Safety Significance

The steam generator tube that ruptured at Ginna exhibited no ECT indication during earlier testing. Improved ECT techniques would most likely have given indications and the event could have been avoided.

Possible Solution

This effort, conducted in parallel with ongoing ASME Code Committee activities, would incorporate updated eddy current inspection procedures in the ASME Boiler and Pressure Vessel Code, Sections V and XI for NDE and ISI, respectively. The improved test procedures would be considered part of the in-service ECT of PWR steam generator tubing.

Priority Determination

In a previous evaluation[756] by the staff, it was determined that improved ECT techniques would provide small reductions in public risk and, therefore, was considered a low priority. It was also concluded that significant reductions in ORE could result from use of improved ECT techniques. The priority ranking based on the ORE reduction potential was medium. Improved ECT would also enhance the certainty that defective or degraded tubes would be identified and removed from service to assure meeting 10 CFR 100 release limits. The latter condition could be argued to classify improved ECT as a licensing issue. In either classification, an economic incentive for use of improved ECT of up to $5M/plant, based on avoided cost of forced outages, could be obtainable. Based on a combination of the above potential benefits, development of improved ECT procedures was recommended as a medium priority principally because of the potential reductions in ORE.

Conclusion

This issue was integrated into the resolution of Issue 135.[1075]

ITEM 67.8.0: DENTING CRITERIA

Description

Historical Background

This item concerned a staff recommendation to develop generic inspection criteria and methods to quantify steam generator tube denting. Operating experience showed that surveillance of steam generator tubes was necessary to identify denting and to take corrective action to mitigate the stress corrosion cracking induced by denting.

Safety Significance

Denting can enhance stress corrosion cracking leading to through-wall cracks and leaks in steam generator tubes. Denting, combined with flow slot `hourglassing,' caused the U-bend stress corrosion cracking that led to the SGTR at Surry Unit 2 in September 1976.

Possible Solution

Development of a generic inspection requirement and criteria for steam generator tube denting will provide assurance that minimum standards for denting are applied uniformly.

Priority Determination

Frequency Estimate

At the time of this evaluation, only one SGTR event was attributed to the denting phenomena in approximately 300 RY of operation. This corresponded to an SGTR frequency of 3×10^{-3}/RY. Therefore, the SGTR contribution to a core-melt frequency of 4.7×10^{-6}/RY contained a contribution of approximately 15% or 7×10^{-7}/RY due to denting.

Consequence Estimate

The PWR Category 4 release of 2.7×10^{6} man-rem was used to estimate the consequences of a core-melt associated with an SGTR. Using the above frequencies, the public risk, annualized over a remaining plant life of 24 years, yielded a public risk of $[(7 \times 10^{-7})(2.7 \times 10^{6})(24)]$ man-rem/plant or 45 man-rem/plant. Based on the assumption that approximately 40 of the operational and planned PWRs (~90 plants) had or will experience denting problems, the total public risk was approximately 1,800 man-rem. Assuming a 30% reduction due to improved denting surveillance criteria resulted in a total public risk reduction of 13.5 man-rem/plant and 540 man-rem for 40 plants.

Cost Estimate

Industry Cost: It was estimated that, as a minimum, with the use of generic denting criteria from the STS, the industry cost benefit would parallel the NRC cost benefit.

NRC Cost: The estimated cost to develop the denting criteria was based on 3 man-months of effort; at $100,000/man-year, this cost was $25,000. The implementation mechanism was assumed to be an STS revision. It was assumed that the denting criteria in the STS would apply to NTOL and CP plants and those operating plants that experienced denting problems. Using the same ratio (40/90) as used in the above risk determination, 40 of the total of 90 plants will require implementation of the STS denting criteria. It was also estimated that development of generic denting criteria would reduce NRC plant-specific review time by 2 man-weeks/plant. The result was a cost saving of (40)(2)($1,920) or $153,600. The net cost benefit to the NRC was approximately $128,600.

Based on the above assumptions, development of generic denting criteria had a total net cost benefit of approximately $250,000.

Value/Impact Assessment

The public risk reduction associated with implementation of generic denting criteria was not significant. The major value in development of these criteria was that it could provide a net cost benefit to the NRC and industry. No negative impacts (adverse changes to existing plant-specific criteria) were assumed in this evaluation.

Conclusion

In consideration of the low potential public risk reduction, development of generic denting criteria was considered a low priority. However, the generic denting criteria provided a small public risk reduction potential and could result in a net cost reduction for the NRC and industry. The issue was addressed in the resolution of Issue 135.[1075]

ITEM 67.9.0: REACTOR COOLANT SYSTEM PRESSURE CONTROL

Description

Historical Background

This item addressed Recommendation 9 of the DL/NRR memorandum[752] and called for a study to determine the need for controlling and reducing RCS pressure during and following an SGTR with emphasis on existing plant systems and equipment. The spectrum of possible initial conditions, RCS thermal-hydraulic conditions, and break sizes were to be considered. The use of the pressurizer auxiliary system was to be explicitly examined since its use could eliminate the necessity to use the pressurizer PORV in cases where forced RCS flow is lost. The study was to address the following objectives: (1) minimizing the primary to secondary leakage through the broken steam generator tube; (2) maximizing control over system pressure; and (3) minimizing the chances of producing voids in the RCS and other complicating effects.

Safety Significance

RCS depressurization following an SGTR is more difficult because of the loss of normal pressurizer spray. RCS fluid contraction, caused by the cool-down from the dumping of secondary-side steam to either the main condenser or to the atmosphere, will result in some reduction in RCS pressure, but other measures must be taken to expeditiously reduce the RCS pressure to the point where primary coolant flow into the damaged steam generator stops.

The pressurizer PORV was used during the Ginna and Prairie Island SGTR events to reduce RCS pressure. However, control of RCS pressure is difficult with the PORV since its use creates an additional loss of coolant. The decrease in RCS pressure can be so rapid that steam voids may be formed in the reactor vessel upper head and at the top of the steam generator U-tubes and may further complicate the RCS depressurization. Void formation can lead to concerns regarding core cooling. The Ginna operators were sufficiently concerned that they left the safety injection pumps operating, thereby overfilling the steam generator via primary-to-secondary leakage through the ruptured tube. The resulting secondary-side pressure transient caused the main steam safety valves to lift, releasing radioactive material directly to the atmosphere. It was not apparent that the auxiliary spray from the charging system could have successfully lowered RCS pressure to the point where primary coolant flow into the steam generators could have been stopped. It may have been that, by spraying cold charging fluid into the pressurizer, the decrease in pressure would have resulted in void formation thus expanding the RCS fluid volume, filling the pressurizer, and rendering further spray flow ineffective. This phenomenon was to be examined as well as the thermal stresses on the spray nozzle.

Possible Solution

With optimized RCS pressure control, the risk associated with an SGTR could be reduced by reducing the potential radiological consequences.

Priority Determination

Frequency Estimate

Independent analyses by the staff considered three categories of SGTR events: (1) SGTR and loss of DHR; (2) SGTR resulting from LOCA; and (3) SGTR with loss of secondary system integrity. For Categories 1 and 2 above, the core-melt probabilities were not dominated by SGTRs and were calculated to be 5.5×10^{-7}/RY and 3×10^{-8}/RY, respectively. Category 3 included single and multiple tube ruptures followed by stuck-open steam generator safety valves, MSLB, failure of the MSIVs, steam generator overfill, and failure to depressurize the RCS before the RWST was exhausted. The latter was considered since recirculation water from the sump might not be available following an SGTR event should a loss of secondary system integrity (e.g., stuck-open safety valve, MSLB) occur outside containment.

It was assumed that RCS pressure control would enhance depressurization of the RCS by a factor of 10 for the Category 3 sequences involving less than 10 SGTRs. For greater than 10 SGTRs, the depressurization was assumed to be too rapid for the RCS pressure control to be effective. The result would be a reduction in core-melt frequency of 1.8×10^{-6}/RY for enhanced RCS pressure control.

Consequence Estimate

The consequences resulting from an SGTR would involve releases typical of a PWR Category 4 release as used in WASH-1400[16] and modified to a typical meteorology with a population density of 340 persons/square-mile within a 50-mile radius of the affected plant. The public risk reduction was $(1.8 \times 10^{-6})(2.7 \times 10^{6})$ man-rem/RY or 4.9 man rem/RY. Considering an average remaining plant life of 24 years, the public risk reduction was estimated to be 117 man-rem/reactor.

Cost Estimate

NRC Cost: The cost of the recommended separate staff study depended on the existing capability for RCS pressure control following an SGTR and the incremental improvement required. As a minimum, the study could require a review and documentation of how existing systems and procedures already provided the requisite capability. In some plants, the study could require thermal-hydraulic modeling of the primary and secondary coolant systems as well as detailed stress analysis of selected components such as the pressurizer auxiliary spray nozzle. A study of this depth and the development of an optimized approach for RCS pressure control could cost one man-year ($100,000) or more.

TMI Action Plan Item I.C.1, clarified in NUREG-0737,[98] included in its scope the development of EOPs for accidents and transients including multiple SGTRs. Likewise, the adequacy of existing and alternate means of satisfying LWR shutdown decay heat removal requirements were addressed in Issue A-45. Shutdown requirements in effect during SGTRs in PWRs were also considered in Issue A-45. Therefore, existing NRC studies negated the need for a separate study on RCS pressure control.

Industry Cost: The major cost of the study, as recommended, would be borne by the NRC and its contractors; however, input by and consultation with specific plants, plant types, or perhaps separate PWR Owners' Groups would be involved. In the latter case, NSSS Owners' Groups evaluated means of controlling reactor coolant pressure during an SGTR. The depth and scope of the Steam Generator Owners' Group (SGOG) study was expected to at least parallel the above NRC study.

The cost of implementing an optimized approach for RCS pressure control was likely to be highly variable, depending on the adequacy of the existing RCS pressure control capability and the differences between the existing and the optimized approach. The cost associated with implementing an optimized approach for RCS pressure control was not quantifiable, but could include some or all of the following items of cost: (1) developing, validating, and implementing new emergency procedures; (2) training plant operators; or (3) replacing equipment or upgrading equipment qualification if existing equipment must be operated outside of the conditions for which it was originally designed and qualified. In the scope of the recommended study, the implementation cost was moot. However, in an overall value/impact, the implementation cost could be significant.

Value/Impact Assessment

The value of the recommended NRC study on RCS pressure control was that it could uncover, or result in development of, optimized means (procedures, equipment, instrumentation) to control reactor coolant pressure to minimize primary to secondary leakage following an SGTR. Thus, the potential for overfilling a steam generator and the quantity of radioactive material released directly to the atmosphere following an SGTR should be reduced.

Based on the above frequency and consequence estimates, the value was a potential public risk reduction of 117 man-rem/reactor over an average remaining plant life of 24 years. The major initial impact was the cost of performing the study. Subsequent impacts depended on the results of the study and could not be quantified.

Conclusion

Based on the above, the potential public risk reduction of 117 man-rem/reactor that could be derived by a separate (new) NRC study on RCS pressure control was not highly significant. The potential value that could result from such a study would most likely be improved RCS pressure control for both accidents and transients. In this regard, staff actions developed under TMI Action Plan Items I.C.1(2,3) and Issue A-45 also resolved the objective of this issue. In addition, the work by the SGOG on RCS pressure control could have been factored into the review of Items I.C.1(2,3) and Issue A-45.

In summary, RCS pressure control was considered part of studies conducted for NUREG-0737,[98] Items I.C.1(2,3), (which were resolved and implemented under MPAs F-04 and F-05) and Issue A-45.

ITEM 67.10.0: SUPPLEMENTAL TUBE INSPECTIONS

Description

Supplemental Tube Inspection (STI) was originally proposed by the staff as a recommended licensee action.[752] The value/impact analysis[756] ranked the proposed staff recommendation as a licensing issue. This ranking inferred that the staff-proposed STI would provide only small potential public risk reduction and a low value/impact ratio. However, as a minimum, the statistical sample size of the proposed STI would ensure that no more than the limiting number of defective tubes would go undetected. The limiting number of sample tubes to be inspected would be based on meeting 10 CFR 100 release limits from, and concurrent with, an MSLB. Thus, STI would provide additional assurance that existing regulatory requirements on radiological releases would be maintained and further reduce SGTRs. Subsequent information[753] from industry indicated that the staff-proposed STI would result in higher costs and greater ORE than that previously estimated by the staff. The staff reevaluated[753] their proposed STI and agreed in part with the industry assessment. However, it was the staff's position that some form of STI could be formulated to provide added assurance of tube integrity with less ORE and an improved value/impact relationship.

In view of the above, STI did not require licensee implementation but was identified for further staff action and evaluation.

Conclusion

This issue was classified as a Licensing Issue that called for the staff to investigate more practical alternatives for STI. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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ISSUE 81: IMPACT OF LOCKED DOORS AND BARRIERS ON PLANT AND PERSONNEL SAFETY

Description

Historical Background

In October 1982, the Executive Director for Operations appointed the Committee to Review Safety Requirements at Power Reactors (CRSRPR) to review U.S. Nuclear Regulatory Commission (NRC) security requirements at nuclear power plants with a view toward evaluating the impact of these requirements on operational safety. Overall, the CRSRPR did not identify any clear operational safety problems associated with implementation of the NRC's security requirements. However, the Committee found that there was the potential for security measures at a site to adversely affect safety and issued its recommendations in a report[621] to the Office of Nuclear Material Safety and Safeguards. In view of one of the findings in this report, a memorandum[542] was issued on May 31, 1983, identifying this issue and suggesting that a multidisciplinary group be convened to perform an integrated assessment of the potential safety problem associated with locked doors and barriers. Based on the responses to the memorandum, a consensus supported the creation of the multidisciplinary group to gather the necessary information and prepare a scope of the issue for appropriate consideration. [623] This approach was approved[624] and action on this matter was formally initiated.[625]

The multidisciplinary group held its first meeting on February 28, 1984, and issued a report on June 8, 1984.[626] Inasmuch as a proposed rule (SECY-83-311, "Proposed Insider Safeguards Rules," dated July 29, 1983[627]) specifically designed to address the security barrier issue had been prepared independently, and IE Information Notice 83-36, "Impact of Security Practices on Safe Operations,"[628] had been issued in June 1983, the work of the group was limited to nonsecurity barriers.

The proposed rule[1436] was eventually adopted and stated that "the NRC is amending its regulations to provide a more safety conscious safeguards system while maintaining current levels of protection." Regulatory changes included (1) permitting suspension of security based on Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(x) and (y), (2) requiring the access authorization system to be designed to accommodate the potential need for rapid ingress and egress of individuals during emergency conditions or situations that could lead to emergency conditions, and (3) ensuring prompt access to vital equipment by periodically reviewing physical security plans for potential impact on plant and personnel safety. The rule was implemented with Regulatory Guide (RG) 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls,"[1438] and Generic Letter 87-08, "Implementation of 10 CFR 73.55 Miscellaneous Amendments and Search Requirements," dated May 11, 1987,[1437] which addressed the issuance of vital area keys to operations personnel. At the time of evaluation of this issue in 1995, the Office of Nuclear Reactor Regulation (NRR), Reactor Safeguards Branch, indicated that almost all licensees were in compliance with RG 5.65[1438] and Generic Letter 87-08[1437] and had implemented mechanical key overrides for electronically controlled access doors. The rulemaking resulted in security plan amendments that increased the focus on plant and personnel safety.

Subsequent to the above work, a main feedwater pipe rupture event at Surry Power Station (see Issue 139, "Thinning of Carbon Steel Piping in LWRs") caused the failure of a security card-reader that was located approximately 50 feet from the break point. This failure was caused by intrusion of water and steam that saturated the card-reader. As a result, key cards could not be used to open plant doors. The control room doors were opened to provide access to the control room, and security personnel were assigned to the control room to provide access security. One operator was temporarily trapped in a stairway due to the card-reader failure. Electric override switches were later installed to remedy this problem. Because of the failure of the security card-reader during the Surry Power Station event, the staff determined that Issue 81 should be expanded to include potential electric door lock failures and reevaluated to determine whether the previous priority ranking (DROP) should be changed.[1163]

Safety Significance

The possible failure of locked doors and barriers that may be required for fire protection, radiation protection, flood protection, and administrative controls is of special concern during abnormal or accident situations when emergency conditions may require prompt and unlimited access of the plant operators to safety equipment to assure proper plant shutdown. This issue was applicable to all operating and future plants.

Possible Solution

An evaluation of each plant's locked doors and barriers might be required and appropriate procedural and hardware changes may have to be made to establish that operator access is unimpeded during emergency, abnormal, or accident conditions, and that prompt operator action, as required, is possible.

Priority Determination

This section presents the NRC staff analysis for prioritizing this issue, which was performed in 1995. This analysis, which includes frequency, consequence, and cost estimates and a value/impact assessment, has not been updated in the 2011 revision of this issue.

In the event of an accident, failure of the electronic card-reader access control system (ACS) could result in an impediment to operator actions outside of the control room that are required for recovery. Some examples of possible operator actions are (1) locally overriding a failed component, (2) replacing or repairing a failed component, or (3) realigning valves to bypass a failed pump or clogged pipe. If the card-reader ACS fails, the operator will be impeded in his access through the door.

Even if the ACS fails, there is a large probability that the plant will have a mechanical key override or that the locks will fail open. The study conducted by the CRSRPR estimated that a majority of plants did not have problems with ACS computer failure, either because the doors fail open, mechanical key overrides are available, or the number of controlled areas is small.[621] An NRR review of plant safeguards revealed that only one plant that did not have a mechanical key override on ACS-controlled doors had locks that failed open. Based on these data, a probability of 0.01 was assumed to account for the occurrence of no key override due to lost or misplaced keys, mechanical failure of the override, or failure of an electronic ACS to fail open if so designed.

Assuming the worst case (i.e., the operator has no other means than to defeat the lock), the effect of impeded operator action was estimated assuming that action begins soon after the accident is initiated. The amount of time between accident initiation and the initiation of core damage was calculated for critical minimum cutsets in WASH-1400 (NUREG-75/014, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," issued October 1975.[16] Studies have also been performed to estimate the amount of time required to defeat a locked door, such as the access delay technology transfer manual (Sandia National Laboratories report SAND87-1926/1 VC-525, dated June 7, 1989). These studies showed that the lock on a typical access door in a commercial nuclear power plant could be defeated in less than 6 minutes with hand tools. The number of doors that would be required to be unlocked can also be estimated. The CRSRPR study found that, on the average, operators need to access three doors to perform routine surveillance, starting from the control room.[621] Routine surveillance requires accessing most areas of the plant, as opposed to the type of specific access that would be required for operator actions in response to equipment failure. Therefore, assuming that three doors will need to be accessed is probably a conservative assumption, even for plants that require more than three controlled doors to be accessed for routine maintenance.

Based on the typical door construction at nuclear power plants, as provided in the Sandia National Laboratories report, the time for penetration of a door with hand tools (e.g., a large screwdriver or crowbar) was determined to range from no delay to 6 minutes. The maximum time required to obtain tools, starting from the control room, was estimated to be 5 minutes. These estimates yield a minimum (5 minutes) and maximum (23 minutes) time for breaking through three doors. These times were arbitrarily assumed to be the 10-percent and 90-percent points on a success probability curve, with linear interpolation between these points and the probabilities of zero-percent chance of success at time = 0 minutes and 99-percent chance of success at time = 1 hour. Using this curve, the probability of successful performance for a given period of time to core damage can be estimated. This curve was used only to estimate the probability that three locked doors can be defeated before core damage is initiated for any given accident sequence, given the unknown construction of locked doors in the average plant and the unknown availability of tools. For this evaluation, every sequence that results in damage in greater than 1 hour was assumed to result in core damage after 1 hour. This was a conservative assumption because almost every sequence requires more than 1 hour until initiation of core damage and/or containment failure. A detailed analysis of each accident scenario would be required to determine at what point a given action would become futile.

In order to estimate the probability that the card-reader will fail during a plant transient or similar event, data from NUREG/CR-5580, “Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment,” issued December 1992,[1588] were reviewed. This report contained details from 138 incidents of advertent or inadvertent fire protection system actuation. Only one incident resulted in failure of the electronic ACS. Based on these data, a failure rate for the ACS of 0.01 was assumed. Although this estimate may be close to the actual failure rate for a scenario like station blackout, it is probably conservative for most other scenarios. (In addition, the harsh environment criterion was considered by NRR in the licensing of new plants, and the advanced boiling-water reactor design incorporates features to prevent the failure at one station from affecting the rest of the system.)

Scenarios that require operator action outside the control room are included in most probabilistic risk assessments (PRAs). The quantification of those scenarios usually includes the probability that the operator will fail to properly respond to an equipment malfunction or other problem in a timely manner. The effect of ACS failure on operator actions can be included in these terms by increasing the operator error probabilities used in the existing PRAs by a factor equivalent to the combined probabilities of card-reader failure, key override unavailability, and operator inability to break through doors before initiation of core damage. These data are presented in Table 3.81-1 below. However, because it is recognized that this table does not include individual sequences that may have a significantly greater chance of impeded operator action, it was decided to perform a parametric study in order to determine the impact on the probability of core damage of the variation in those estimates. The results of this study are presented in Table 3.81-2 and discussed below. In addition, a review of the interfacing systems loss-of-coolant accident (LOCA) accident scenarios for three typical plants (which were investigated in a separate program) indicated that only one of the three plants had a minimum cutset that included an operator action outside of the control room. This action also had an alternate action inside of the control room.

Frequency Estimate

In order to calculate the increase in core-melt frequency, the minimum cutsets from existing PRAs for Oconee Nuclear Station, Unit 3, and Grand Gulf Nuclear Station, Unit 1, were assumed to represent the ideal condition of no effect of locked doors on operator recovery for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs), respectively. This was a reasonable assumption because the scenario of card-reader failure was probably not considered for these early PRAs and was not discussed in the Nuclear Safety Analysis Center reports on these PRAs.

The effect of the card-reader ACS failure was studied[64] parametrically by Pacific Northwest Laboratory by calculating the total increase in core damage frequency (CDF) for a corresponding increase in the probability of any event that models failure of operator action. This probability of operator failure was increased by 0.0001, 0.0005, 0.001, 0.01, and 0.1. This quantity was added to all existing parameters in minimum cutsets that represented failure of necessary operator actions outside of the control room. These actions are included in all minimum cutsets, not only the four listed in Table 3.81-1. The results of this study are presented in Table 3.81-2.

Although the maximum probability calculated in Table 3.81-1 is 1.2×10^{-5}, the range of values for the parametric study started at 10^{-4}. However, 1.2×10^{-5} was believed to be a conservative estimate of the maximum increase in operator failure probability, as the events listed in Table 3.81-1 were chosen on the basis of a high speed of core damage initiation. Events that take longer than several hours to develop core damage will probably be minimally affected by impeded operator access, because the operators will have more time to gain access. Therefore, including these events in the calculation for increase in CDF was conservative.

Consequence Estimate

The increase in public risk was an output of the PRA and was also shown in Table 3.81-2. Corresponding values for total public exposure were calculated based on the estimated number of operating and future plants (90 PWRs and 44 BWRs) with remaining lives of 28.8 and 27.4 years, respectively. As expected, the increase in both core-melt frequency and public risk was negligible at the expected levels of operator impairment (10^{-4}) but became significant at unrealistic levels of impairment (10^{-1}). These calculations contain the implicit assumption that core damage will occur in no more than 1 hour for all events.

Table 3.81-1 Estimated Probability of ACS Failure To Prevent Operator Action

Event[d]	Probability			Product[c]
	ACS Fails[a]	Override Fails[a]	Delay Exceeds Limits[b]	
V	0.01	0.01	0.12	1.2×10^{-5}
S_2D	0.01	0.01	0.08	0.8×10^{-5}
AH	0.01	0.01	0.02	0.2×10^{-5}
TMLB	0.01	0.01	0.01	0.1×10^{-5}

Notes

[a] Estimated.

[b] Calculated, assuming a limit of 1 hour for all sequences.

[c] Based on assumed independence of ACS failure, override failure, and delay of operator until core damage initiates.

[d] Minimum cutset accident sequences from WASH-1400:[16]

V LPIS check valve/system failure
S_2D 0.5" to 2.0" LOCA combined with loss of ECCS injection
AH Medium to large LOCA and failure of ECCS recirculation
TMLB TMI sequence

Table 3.81-2 Calculated Increase in Core-Melt Frequency and Public Exposure

Operator Failure Probability[a]	Core-Melt Frequency (x 10^{-5}/RY)				Public Dose Increase (man-rem)
	PWR	Δ	BWR	Δ	
Base Case	1.408	-	2.475	-	-
0.0001	1.411	0.003	2.482	0.007	5.2×10^2
0.0005	1.425	0.017	2.509	0.034	2.6×10^3
0.001	1.442	0.034	2.543	0.068	5.2×10^3
0.01	1.754	0.35	3.149	0.674	5.2×10^4
0.1	4.968	3.6	9.216	6.74	5.2×10^5

Note
[a] Increase in probability that operator will fail to perform recovery action within the necessary time due to card-reader ACS failure and locked doors.

Cost Estimate

Based on the deliberations of the multidisciplinary group, the cost to evaluate and make modifications to each plant and its procedures was estimated to be approximately $1.1 million (M) per plant.[626] This cost was based on the following factors:

(1)	A one-time evaluation of existing plant locked doors and barriers	$ 200,000
(2)	Resolution of adverse safety findings (Cost for maintaining keys for a security force of 24 per plant was estimated to be $21,000/reader.[627] Training for security and operational personnel based on 50 operators and security personnel for 1 day/year/plant, over the lifetime of the plant (28 years) was assumed to be (1/365)(50)(28)($100,000) = $391,232)	400,000
(3)	Ongoing program to ensure future reduction of safeguards impact on safety ($10,000/year for an average reactor lifetime of 28 years)	280,000
(4)	NRC reviews of plant modifications	200,000
	TOTAL:	$1,080,000

These estimates could be high for a plant that was in substantial compliance with the recommendations in Generic Letter 87-08[1437] and RG 5.65.[1438] However, because the estimated safety benefit for these plants would be a decrease in CDF significantly less than 10^{-5}, these plants would not meet the substantial additional protection criterion of the Backfit Rule (10 CFR 50.109, “Backfitting”).

Value/Impact Assessment

The value/impact assessment is presented in Table 3.81-3.

Table 3.81-3 Value/Impact Assessment

Probability Increase	Total Cost ($M)	Risk Reduction (man-rem)	S (man-rem/$M)	Priority Ranking
0.0001	150	$5.2x10^2$	3.5	DROP/LOW
0.0005	150	$2.6x10^3$	17	LOW
0.001	150	$5.2x10^3$	35	LOW/MEDIUM
0.01	150	$5.2x10^4$	350	MEDIUM/HIGH
0.1	150	$5.2x10^5$	3,500	HIGH

Other Considerations

The following other considerations relate to this issue:

(1) The most probable effect of locked doors on reactor safety was believed to be represented by an increase in the probability of failure of the operator to leave the control room and perform actions required for recovery of less than 0.0001. This corresponded to a priority rating that was borderline between DROP and LOW priority. Even if this estimate was inaccurate by an order of magnitude, the corresponding priority ranking would be borderline between LOW and MEDIUM.

(2) Even with a conservative assumption about the impact of failure of the ACS on the probability of preventing operator recovery action, the issue would not satisfy the requirements of the Backfit Rule (10 CFR 50.109). Specifically, SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," dated August 27, 1991,[1425] stated that, with limited exceptions, a reduction of CDF of at least 10^{-5} was needed to satisfy the substantial additional protection criterion of that rule. However, Table 3.81-2 shows that, with the probability of operator failure due to ACS failure as high as 10^{-2}, the change in core-melt frequency does not reach this value. Further, as shown in Table 3.81-1, the best estimate of the increase in the probability of operator failure is in the range of 10^{-5}.

(3) This evaluation was not intended to address the effect of locked doors on worker safety in an operating plant. A nuclear power plant has many inherently dangerous materials that may present a significant hazard to untrained personnel but do not significantly affect the ability of the plant to safely shut down in the event of an accident or transient. While it was recognized that these dangers pose legitimate concerns, it is beyond the authority of the NRC to regulate working conditions other than radiological hazards.

(4) The consequence and cost estimates described above were based on a remaining life of 28.8 years and 27.4 years for PWRs and BWRs, respectively, consistent with the original 40-year license period. If it were assumed that 75 percent of the plants will have their licenses extended for an additional 20 years, the remaining life would be increased by 15 years. This would have very little impact on the value/impact assessment described above.

Conclusion

As explained above, this issue was initially placed in the DROP category in 1984. The estimated frequency of card-reader ACS failure and its impact on plant safety indicated that improvements in this area were not a cost-effective way to increase overall plant safety. Moreover, the multidisciplinary task group concluded that the locks and barriers associated with these areas could easily be defeated or bypassed in an emergency situation, if necessary, provided there was enough time to take the necessary steps. In addition, implementation of the regulatory guidance associated with rulemaking[1436] resulted in better coordination between plant security and operations personnel. Thus, this issue was given a LOW priority ranking in 1992 (See Appendix C). Consideration of a 20-year license renewal period did not change the priority of the issue.[1564]

The staff conducted a review of this issue in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluation.[1964] The staff determined that the operating experience has not indicated a change in the significance of this issue. In addition, the staff verified that the regulations related to this issue establish requirements that provide prompt access to affected areas and equipment during emergencies. The following discussion demonstrates the application of the NRC regulatory framework to this issue.

According to 10 CFR 73.55(e)(9)(i), “Vital equipment must be located only within vital areas, which must be located within a protected area so that access to vital equipment requires passage through at least two physical barriers, except as otherwise approved by the Commission and identified in the security plans.” During emergencies or abnormal conditions, it may be necessary for certain licensee personnel to gain quick access to vital equipment to mitigate or terminate some adverse plant condition. The regulation at 10 CFR 73.55(g)(5)(i) requires that “The licensee shall design the access control system to accommodate the potential need for rapid ingress or egress of authorized individuals during emergency conditions or situations that could lead to emergency conditions.” Moreover, 10 CFR 73.55(g)(5)(ii) states that “To satisfy the design criteria of paragraph (g)(5)(i) of this section during emergency conditions, the licensee shall implement security procedures to ensure that authorized emergency personnel are provided prompt access to affected areas and equipment.”

In addition, requirements have been established to ensure that personnel can quickly evacuate vital areas if the emergency condition results in high radiation or other dangerous conditions within the vital area. The regulations at 10 CFR 73.55(e)(8)(iii) and 10 CFR 73.55(e)(9)(ii) state, in part, this requirement for protected areas and vital areas, respectively. The regulation at 10 CFR 73.55(e)(8)(iii) states that “All emergency exits in the protected area must be alarmed and secured by locking devices that allow prompt egress during an emergency and satisfy the requirements of this section for access control into the protected area.” In addition, 10 CFR 73.55(e)(9)(ii) states that “The licensee shall protect all vital area access portals and vital area emergency exits with intrusion detection equipment and locking devices that allow rapid egress during an emergency and satisfy the vital area entry control requirements of this section.”

Finally, Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” states that administrative controls shall establish procedures to define the strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. Under these strategies, in part, “All access and egress routes that involve locked doors should be specifically identified in the procedure with the appropriate precautions and methods for access specified.”

In addition to the regulations stated above, for emergencies or abnormal conditions, RG 5.65[1438] states that “Licensees can provide for rapid ingress/egress during such conditions by providing backup keys to vital areas and methods of opening locked doors in the case of computer or power failure." Moreover, RG 5.65[1438] describes acceptable procedures for providing for safe ingress/egress during a power or computer outage.

Based on the review of the NRC regulations related to this issue presented above, the staff concluded that the existing regulations adequately establish requirements that provide prompt access to affected areas and equipment during emergencies. Therefore, the staff changed the status of Generic Issue 81 and DROPPED this issue from further pursuit.

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1564. Memorandum for W. Russell from E. Beckjord, "License Renewal Implications of Generic Safety Issues (GSIs) Prioritized and/or Resolved Between October 1990 and March 1994," May 5, 1994. [9406170365]

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ISSUE 111: STRESS CORROSION CRACKING OF PRESSURE BOUNDARY FERRITIC STEELS IN SELECTED ENVIRONMENTS

Description

Historical Background

Indications of possible stress corrosion cracking (SCC) in the Indian Point Unit 3 (IP-3) steam generator prompted MTEB to review foreign and domestic operating experiences related to possible indications of SCC in low-alloy ferritic steels. The incidents identified[837] as possible precursors to generic concerns of SCC relate to BWR reactor vessels and PWR steam generators. These events and some additional information that are reviewed and discussed in this evaluation include:

(1) A through-wall crack in the transition cone of the steam generator shell at IP-3.

(2) A through-wall crack in the lower head closure weld region of the Italian Garigliano steam generator (an indirect cycle BWR similar to a BWR-1).

(3) A guillotine rupture of a transition cone (reducer) in the secondary piping of the German HDR test facility.

(4) Cracking of feedwater lines in W PWRs.

(5) Other events that may contribute to SCC in BWR reactor vessels and PWR steam generator vessels.

(6) Inferences from materials testing.

The materials of interest are those low-alloy ferritic materials (SA-533 Grade B, SA-508 Grade 2, and SA-302 Grade B) used in the fabrication of the subject pressure vessels.

Safety Significance

The reactor vessels and steam generators are constructed of low-alloy ferritic steels and designed to the ASME Codes. The ASME Codes are linked to fatigue crack initiation in chemically unreactive environments (ASME Section III) and fatigue crack growths of existing defects as part of the ASME Section XI inspection Code. Even though a corrosion allowance is specified in the ASME Codes as a design consideration, it is not linked to corrosion fatigue or SCC that may occur in active chemical environments such as those experienced in the nuclear pressure vessels (reactor pressure vessels, steam generator pressure vessels).

Should the materials used in the pressure vessels be susceptible to SCC and exceed the inherent allowances in the ASME design/inspection Codes, a vessel rupture could result in a core-melt and radiation doses to the public. This issue affects the design and operation of all LWRs except those designed by B&W.[859]

Possible Solution

Prior to developing a solution to this problem, MTEB proposed a research scoping effort to define the severity of the problem and the conditions under which the SCC phenomena are likely to be exacerbated. The research effort would also involve laboratory testing of the low-alloy materials in reactor-grade water with variable oxygen, chloride, and copper as possible water chemistry constituents.

No risk reduction can be attributed to the study (scoping) efforts. However, the proposed effort is expected to better define under what conditions SCC of the pressure boundary steels may occur and if such conditions arise or prevail during reactor operations. The proposed effort would also involve determinations of the effectiveness of post-weld heat treatments (PWHT) and water chemistry excursions that may affect the materials resistance to SCC. The results of these studies (research) could then possibly be used to determine when and where to conduct inspections to detect the cracks before they become a safety concern.

Priority Determination

In order to develop background frequency information to establish the safety significance of this issue, a review and discussion of the incidents identified above was required.

IP-3 Steam Generator Event: During a refueling outage (with the reactor in a cold shutdown condition) on March 27, 1982, a small leak was detected on the shell side of steam generator #32 of IP-3. The leak originated in the circumferential weld joining the transition cone to the upper shell. The steam generator shell is constructed of SA-302 Grade B material approximately 4 in. thick. To characterize the cracking phenomenon, the utility had various samples removed for metallurgical evaluation and failure analyses. BNL performed an independent failure analysis on specimens from steam generator #32 and on three additional boat samples containing cracks cut from steam generator #31. Office of Inspection and Enforcement (OIE) issued Information Notice No. 82-37[842] to inform the industry of the event. W informed[843] the NRC staff that no indications similar to those observed at IP-3 were identified in the inspections performed on steam generators in 12 plants.

An investigation by BNL as reported in NUREG/CR-3281[844] concluded that the cracking was caused by a low cycle corrosion fatigue phenomenon with cracks initiating at areas of localized corrosion (pits) and propagating by fatigue. The cause of the pitting/cracking was considered to be related to the unit's relatively high operating dissolved oxygen levels and copper species in solution. The report also concluded that SCC could not be entirely discounted as the possible failure mechanism. NUREG/CR-3281[844] also identified that IP-3 had developed moderate to severe denting of the steam generator tubes. The sludge analysis in IP-3 showed concentrations as high as 45% copper and 40% iron. Significant amounts of chlorine (Cl), copper as cuprous oxide (Cu_20), and alpha hematite (alpha-Fe_20_3) were also present in the sludge pile. The presence of these constituents indicated that water chemistry control in the IP-3 steam generators had been poor for a considerable period of time. Additionally, in January 1981, IP-3 experienced a turbine blade failure which damaged approximately 50 condenser tubes and allowed chloride into the steam generators with recorded levels of up to 325 ppm. The chloride intrusion may have had some influence in initiating pits at the inside surface of the steam generator shell.

Results from constant extension rate tests (CERT) on SA-302 Grade B material in neutral and chloride solutions were reported in NUREG/CR-3614.[845] The CERT were performed on weld and base metal samples in air, water, and chlorine solutions. The chlorine solutions as sodium chloride (NaCl) and cupric chloride ($CuCl_2$) ranged from 1 ppm to 325 ppm chlorine. The results of the test indicated no significant effect in the NaCl CERT. However, the $CuCl_2$ CERT indicated possible susceptibility of the SA-302 Grade B material with as little as 1 ppm chlorine (as $CuCl_2$) in 268°C water. No attempt was made to control the dissolved oxygen content in the water. The combined results appear to indicate that copper as $CuCl_2$ may significantly alter the electrochemical reaction. The IP-3 secondary water chemistry may, however, provide an even different corrosion mechanism than that of the CERT. In this regard, the electromotive force series of metals could also produce galvanic corrosion of the iron (Fe) in the presence of copper because carbon steel is anodic compared to copper (Cu) in the galvanic series. Thus, pitting/crevice corrosion of the carbon steel may have been acting as a combination of galvanic corrosion and low cycle fatigue. In the latter case, corrosion products in cracks (crevices) may act as wedges during cooldowns causing crack extensions. During heatups, newly-exposed crack surfaces develop more corrosion deposits. Repeated cycles, therefore, may result in through-wall cracks (corrosion-fatigue).

Because of the poor secondary water chemistry control at IP-3, the atypical massive chloride intrusion, and the results of the W inspections on other steam generators, the event at IP-3 may not represent a generic PWR condition but a plant-specific combination of atypical events. However, because of uncertainties in the CERT to represent conditions that may have prevailed at IP-3, and the indications from the CERT of the potential for copper in solution to effect some form of corrosion-related attack on the low-alloy materials, these effects cannot be ruled out as a potential generic concern, especially when considering the PWR secondary water chemistry controls that have existed in the industry (see "Other Conditions" contributing to SCC).

Garigliano Steam Generator Event: The Garigliano steam generator crack developed at the inner surface of the water box circumferential weld between the tube sheet and the nozzles on the primary side (August 1978). The through-wall crack propagated through the Monel clad and the SA-302 Grade B shell (approximately 2 inches thick). GE conducted an extensive investigation and reported[846] its results to the NRC. The most pertinent information revealed that the crack propagation resulted from environmentally-assisted corrosion under sustained loads (SCC). Manganese sulfide as segregates were evident in the monel and base metal with the presence of sulfur in the region of crack tips. Therefore, aggressive acidic crack-tip chemistry caused by dissolution of the sulfide inclusions were concluded by GE to be contributors to the SCC. Local PWHT of the weld with unknown control was also reported by GE to have resulted in high residual stresses in the region of the weld. The high oxygen content (~200 ppb) in the coolant medium was not considered atypical, but it may have enhanced the electrochemical reaction involved in the crack initiation and propagation.

GE concluded that the conditions that prevailed in the Garigliano steam generator (high residual stress, material sulfur content and inclusions) were atypical of current domestic BWR design and PWHT. The NRC staff did not challenge the GE position. Therefore, the Garigliano event was not considered a generic event typical to domestic operating BWRs. However, the effects of sulfur content in the material and the potential contribution to SCC have since been subject to further tests and evaluations (see discussion on material testing). One might argue in hindsight that the Garigliano event could have been a precursor to the SCC susceptibility of high/low sulfur content low-alloy steels in reactor grade water.

HDR Rupture Event: NUREG-1061,[611] Volume 3, describes the double-end guillotine break that occurred in the HDR test facility on November 3, 1983. The reducer (conic section) that failed was fabricated from a single billet of 15 Mo 3 steel. The wall thickness of the conic section was approximately one-fourth the design thickness. Therefore, the combined primary, secondary, bending, and notch stress concentrations could have resulted in a stress intensity of nearly two orders of magnitude above the design stress. This fabrication error could well have resulted in exceedance of some stress threshold that caused the failure. The thinness of the conic wall section and the high oxygen content (~8ppm) may also have contributed to the failure. The atypical design and fabrication errors related to the HDR failure are believed sufficient to preclude this event as representative support of this issue as a generic issue. It should be pointed out that, although the stresses were very high, there was no gross plastic deformation and no ductility exhibited on a microscale.[857] It was a brittle fracture. The failure is atypical of fatigue in that there were numerous initiation sites. These facts point to stress corrosion cracking of low alloy/ carbon steels as the failure mechanism. This incident is cited to demonstrate the mechanism.

PWR Feedwater Line Cracking Events: These failures are being addressed in Issue 14. The primary failure mode has been identified as thermal fatigue (not CF or SCC) resulting from coolant stratification. The PWR Pipe Crack Study Group completed its investigation of this issue and published its findings in NUREG-0691.[13] Based on the above findings, any SCC that may or may not have influenced the resulting failures were masked by the thermal fatigue constituent.

Other Events Contributing to Potential SSC: Intrusion of chloride, sulfide, copper, and other contaminants into the BWR reactor water and PWR secondary water may contribute to SCC of the vessels materials. EPRI NP-1136[847] stated that 20 BWR plants over a 33-month time period (1974-1977) indicated 12 forced outages as a result of high conductivity in the reactor water or heavy condenser tube leakages. On an average, this amounts to 307.22 significant contaminant intrusions per BWR reactor-year. EPRI NP-2230[307] reported 6 condenser leakages over 172 RY of PWR operation. This amounts to a frequency of 0.03 contaminant intrusions from condenser leaks into the PWR secondary cooling water of the steam generators.

As a further example of other apparent poor PWR secondary water chemistry operations (in addition to the IP-3 sludge analyses discussed earlier), the sludge deposits in the removed Surry 2A steam generator undergoing tests at Hanford were reported in NUREG/CR-3842.[849] Analyses of the Surry sludge deposits revealed 35 to 60 percent metallic copper, 20 to 30 percent Hematite (Fe_20_3), and 10 percent Cuprite (Cu_20). All the analytical data on the sludge samples indicated that they originated from the secondary side. The high copper content probably originating from the condenser tubing (see "Other Considerations").

Tighter requirements for reactor water may account for the reported higher frequency of contaminant intrusion in BWRs from condenser tube leaks. However, Regulatory Guide 1.56[848] provides methods determined acceptable by the NRC staff to maintain high purity water in the BWR reactor water cycles and to minimize failure of the reactor vessel from mechanisms of general corrosion and SCC induced by impurities in the reactor coolant.

For the secondary side of the PWRs, resolution[850] of Issues A-3, A-4, and A-5 contained staff recommendations that the PWR plants incorporate Revision 3 to SRP[11] Section 5.4.2.1 as plant-specific programs for secondary water chemistry control.

From the above limited data, condenser tube leaks in BWRs and PWRs have been frequent. However, the water purity requirements for BWR plants should alleviate potential corrosion effects to the BWR reactor vessels. For the PWR steam generators, adoption of the secondary water chemistry guidelines may reduce future corrosion potentials, but not necessarily resolve the effects of existing corrosion damage.

Based on the IP-3 experience, the above-described Surry sludge analyses, the recent Surry Unit 2 inspections discussed in "Other Considerations," and the fact that steam generator tube degradations have been linked to variable PWR secondary water chemistry controls,[840] it appears reasonable to equate the adequacy of the steam generator secondary water chemistry environment to conditions that may also enhance SCC in the steam generator vessel shells.

Inferences from Materials Testing: A considerable amount of materials research and testing has been performed on the SA-508 and SA-533 reactor vessel materials and has resulted in the publication of several documents: NUREG/CP-0058,[851] Vol. 4; NUREG/CP-0044,[852] Vol. 1 (pp. 7, 91, 141, 179); NUREG/CP-0044,[852] Vol. 2 (pp. 27, 91); Reference 853; and NUREG/CR-4121.[854]

The research and testing were performed in typical PWR and BWR reactor water chemistries. The research results also included comparisons with the ASME Section XI air and water fault lines. Based on the existing research results, the following generalizations appear appropriate for these materials:

(1) There is a trend toward increased crack growth rate with higher material sulfur content.

(2) A higher dissolved oxygen content results in higher initial crack growth rate, but the crack growth rate is stifled with crack depth such that after an initial period of crack growth rate the effects of the bulk solution dissolved oxygen content diminishes. Therefore, there is little difference in the effective crack growth rates of these materials in BWR and PWR reactor water chemistries.

(3) The crack growth rates for reactor pressure vessel materials are within, or consistent with, the ASME Section XI surface (wet) fault lines.

The most significant effect observed was the high/low sulfur content (material variability), and not the oxygen content (environmental variability). The aqueous solutions used in the referenced research did not contain copper in solution, but some tests did contain small amounts of chlorine in solution.

The only research test results obtained for the SA-302 Grade B base material and associated weld material are reported in NUREG/CR-3281[844] and NUREG/CR-3614.[845] These results were discussed in the earlier IP-3 comparisons.

Based on the above discussions, the differences in the dissolved oxygen contents for the BWR and PWR reactor water chemistries are estimated to have little or no effect on the probability of increased crack growth rates for the reactor pressure vessels. Only limited information was available for the (SA-302 Grade B) pressure vessel material. In the presence of the simulated and degraded PWR secondary water chemistry, the SA-302 material may be susceptible to some form of accelerated corrosion attack.

Frequency/Consequence Estimate

BWR Reactor Pressure Vessel Rupture Frequency Estimate: A nominal base case pressure vessel rupture frequency of 10^{-7}/RY is assumed reasonable for the BWR reactor vessels.[16] In consideration of (1) research results of the reactor vessels materials in their respective reactor water chemistry environments, the vessel materials crack growth rates are within the ASME code limits, (2) the protective corrosion shield provided by the cladding on the inside surface of the reactor vessels, and (3) the BWR reactor water chemistry requirements described earlier, no significant increase in the BWR reactor vessel rupture frequency from SCC is anticipated. However, to provide a coarse estimate, it is assumed that a 25% increase in the BWR reactor vessel rupture frequency can be attributed to SCC. This potential increase in BWR reactor vessel rupture frequency is based on the percentage of stainless steel pipe ruptures attributed to SCC reported in NUREG-1061,[611] Volume 1. Because of the observed prominence of SSC in stainless steel pipes, it seems unlikely that the percentage of reactor pressure vessel ruptures due to SCC would exceed 25% of the total vessel rupture frequency without prior history of this condition. The change in BWR reactor pressure vessel rupture frequency that may be attributed to SCC is therefore estimated to be 2.5×10^{-8}/RY.

<u>BWR Consequence Estimate</u>: Assuming that SCC provides a potential change in the BWR reactor vessel rupture frequency (2.5×10^{-8}/RY), the probabilities of radioactive releases in BWR categories 2 and 3, as described in WASH-1400,[16] are 0.1 and 0.9, respectively. Assuming a 1120 MWe BWR, meteorology typical of the Braidwood site, and a surrounding uniform population density of 340 persons per square mile, the public radioactive risk within a 50-mile radius is 0.113 man-rem/RY. Considering a remaining reactor life of approximately 30 years, the public risk is 3.5 man-rem/reactor.

<u>PWR Frequency and Consequence Analyses</u>: A leak or rupture of a single steam generator would likely produce a rapid cooldown of the reactor similar to an inadvertent full-opening of the turbine bypass valves or a main steam line break.[16] The containments are capable of sustaining a complete blowdown of a steam generator. Therefore, rupture of a single steam generator with no additional failures has no significant risk to the public from core-melt or radioactive releases through containment failures. The plant operations and operation responses to such an event are assumed similar to those described in Issue A-22 for a steamline break inside containment. In addition, subsequent and detailed staff evaluations on PWR responses to MSLB with concurrent SGTRs and SBLOCAs were reported in NUREG-0937[860] which concluded that a MSLB inside containment (similar to a steam generator rupture) would likely be bounded by the FSAR analyses and not result in a core-melt.

For a steam generator rupture to lead to a significant release (core-melt), the rupture must be accompanied by damage to the RCS and failure of the ECCS, or failure of the AFWS and the ECCS. The following sections will address these PWR systemic events.

<u>PWR Steam Generator Rupture (SGR) Frequency Estimate</u>: WASH-1400[16] estimated that the SGR frequency was similar to the RPV rupture frequency (10^{-7}/year). Considering approximately 3 steam generators per reactor, the base case SGR frequency is 3×10^{-7}/RY.

To assess the potential increase in SRG frequency as a result of accelerated SCC or CF from PWR secondary water chemistry variability between plants, we reason the following: (1) plants with clean secondary water chemistry will have an SGR frequency equal to the above base case rupture frequency (3×10^{-7}/RY), (2) plants that have experienced medium degradations of the steam generator tubes will have an SGR frequency one order of magnitude greater (3×10^{-6}/RY) than the base case, (3) plants that have experienced severe degradations of the steam generator tubes will have an SGR frequency two orders of magnitude (3×10^{-5}/RY) greater than the base case rupture frequency of 3×10^{-7}/RY.

The above SGR frequency (3×10^{-5}/RY) is back-calculated to estimate the number of steam generator leaks that have occurred by using the piping leak-before-break ratio of 20.[16] The predicted number of leaks based on the above reasoning is (3×10^{-5}/RY)(500 RY)(20) ~ 0.3. Likewise, if we estimate that steam generator ISI has a 10% chance of not detecting cracks in the steam generators before they develop into leaks, 3 steam generators with cracks could be expected. Compared to the 7 steam generators where cracking has been detected, the above crude estimates are fairly good, but a better correlation with leaks and cracks would be obtained from an SGR frequency of 10^{-4}/RY. For comparative purposes, the probability of a MSLB is also 10^{-4}/RY.

Alternately noting that no rupture has occurred in 1500 steam generator years (500 RY) and ignoring the current steam generator ISI experiences for leak-to-crack detection (1/7) and leak-before-break experiences in U.S. and foreign plants (2 leaks with no ruptures), we would estimate an SGR frequency of 10^{-3}/RY. The SGR frequency of 10^{-3}/RY therefore represents a bounding but prudent estimate. Ignoring the ISI crack detection capability and leak-before-break experiences appears prudent because of the uncertainties in estimating these early warning indicators. As an example of the conservatism of ignoring the crack detection capability, a very conservative staff fracture mechanics analysis[839] estimated that a catastrophic rupture of the steam generator would only be predicted to occur from a complete circumferential crack (360°), with a crack depth approaching one-half the vessel wall thickness. A crack of this magnitude seems very likely to be detectable. Therefore, the SGR frequency may range from a best estimate value of 10^{-4}/RY to an upper bound estimate of 10^{-3}/RY.

<u>PWR Steam Generator Support (SGS) Failure and LOCA Frequencies</u>: If cracks develop in the steam generator vessel shells, it was independently judged[16,859] that the steam generator would likely leak before rupture. The SGR event would therefore most likely be bounded by the MSLB event previously discussed. However, should a catastrophic SGR occur, the steam generator reaction loading to the SGS structure is highly uncertain. In recognition of this, we will assume the conditional failure probability of 0.5 for the SGS (SGS/SGR). The SGS/SGR = 0.5 infers that the SGS is as likely to fail as not to fail. Given failure of the SGS, we assume the conditional probability of a large break LOCA (LBLOCA), given a SGS failure, is 1.

<u>PWR Core-Melt Frequencies</u>: The systemic events that are assumed to lead to core-melt conditions as a result of a catastrophic SGR are: (1) damage to the RCS (LBLOCA), and (2) failure of the ECCS in the unaffected loops, or failure of the AFWS and the ECCS in the unaffected loops. The estimated upper bound core-melt frequencies for these sequences are as follows:

Failure Event	**Frequency/RY**
SGR	10^{-3}
SGS/SGR	5×10^{-1}
LBLOCA/SGS	1
ECCS Failure	$\mathbf{10^{-2}}$
	$\Sigma = 5 \times 10^{-6}$

Failure Event	**Frequency/RY**
SGR	10^{-3}
AFWS Failure	4×10^{-5}
ECCS Failure	$\mathbf{10^{-2}}$
	$\Sigma = 4 \times 10^{-10}$ (negligible)

PWR Containment Failure Matrix: Containment response to a core-melt accident from the above LBLOCA/SGR can be grouped into separate plant damage states (PDS). The PDS depends on: (1) the availability of equipment or systems to reduce containment temperature and pressure; and/or (2) containment bypass or failure to isolate containment. The PDS descriptions and probabilities resulting from the LBLOCA/SGR are as follows:

Plant Damage State (PDS)		
PDS	**Description**	**Probability**
A	No containment heat removal or containment sprays	10^{-3} (Reference 16)
B	Containment heat removal and containment sprays available	0.998 10^{-3}
V/B	Given B, but containment bypass through failed MSIVs in ruptured steam generator steam line	(Reference 681)

The containment failure modes are similar to those used in WASH-1400.[16] The conditional probability of the containment failure mode for each PDS is shown in the table below:

Conditional Containment Failure Mode[a]					
PDS	α	δ	β_4	β_5	V
A	10^{-2}	0.96	10^{-2}	-	-
B	10^{-2}	-	-	10^{-2}	-
V/B	-	-	-	-	10^{-3}

a - α, δ, β, V are the containment failure mode conditional probabilities for missile damage, overpressurization, failure to isolate, and bypass, respectively.

The probability of an α failure mode ($\alpha = 10^{-2}$) from an SGR refers to direct containment failure by missile penetration. For a LBLOCA-induced core-melt, the in-reactor-vessel steam explosion has a probability of 10^{-4} to produce a missile that breaches containment. For purposes of this analysis, the α failure mode probability from missiles generated by the SGR is assumed to be 100 times greater than that from an in-reactor-vessel steam explosion. Therefore, even through an in-reactor-vessel steam explosion is likely to occur from a core-melt, its contribution to containment failure is negligible. The corresponding WASH-1400[16] α release category is a Category 1 release due to the containment failure from a missile generated by the SGR.

Steam produced from the SGR by reactor molten fuel (core-melt) and water in the reactor cavity can fail the containment by overpressurization (δ). This would occur only when containment cooling is lost.[16,860] The probability of overpressurization due to hydrogen burn is assumed negligible because the steam concentration in containment will tend to suppress hydrogen burn propagation. The probabilities of the δ mode failures for PDS A and B are assumed to be 0.96 and zero, respectively. The corresponding WASH-1400[16] release for PDS A and B are Category 2 and Category 3, respectively.

Failure to isolate containment (β failure mode) is assumed to have a probability of 0.01. The β4 mode is with containment sprays unavailable and the β5 mode is with containment sprays available. The corresponding WASH-1400[16] release categories for β4 and β5 are Category 4 and Category 5, respectively. The "V" failure mode probability[681] of 0.001 represent containment bypass through the ruptured steam lines in the affected loop with the MSIVs failed open. The conditional PDS = V/B assumes containment sprays are available and the corresponding WASH-1400[16] release category is a Category 3 release.

The basemat melt-through failure mode is a relatively benign failure mode and, with the most likely case of the containment sprays being available, we assume basemat melt-through is precluded.

The LBLOCA assumed to be induced by the SGR may also be accompanied by SGTRs in the affected loop. However, the conditional SGTRs would be dominated by the probability and consequences of the LBLOCA sequences.

PWR Risk Consequences: The PWR risk consequences for a core-melt frequency (5×10^{-6}/RY) resulting from a SGR-induced LBLOCA is 0.4 man-rem/RY. Over a remaining plant life of 30 years, the public risk is 12 man-rem/reactor. The tabulations of the calculated public risk parameters are:

Public Risk Parameters

WASH-1400[16] Release Category	**Containment Failure Mode**	**Release Frequency $(RY)^{-1}$**	**Conditional Dose/Release (man-rem)**	**Public Risk (man-rem/RY)**
1	α	5×10^{-8}	5.4×10^{6}	0.3
2	δ	5×10^{-9}	4.8×10^{6}	0.02
3	V	5×10^{-9}	5.4×10^{6}	0.03
4	β_4	5×10^{-11}	2.7×10^{6}	-
5	β_5	5×10^{-8}	1.0×10^{6}	0.05
Total	-	1×10^{-7}	-	0.4

The release categories and corresponding containment failure modes are described in the Containment Matrix Section above. The release frequencies (Column 3) are the products of the core-melt frequency (5×10^{-6}/RY) and the summed products of the PDS and the conditional containment failure mode probabilities for each PDS provided in the Containment Matrix Section above. The conditional dose (Column 4) is the man-rem per release for each release category. These release doses are based on the fission product inventory of a 1120 MWe PWR, meteorology typical of the Byron site, and a surrounding uniform population density of 340 persons per square mile over a 50-mile radius from the plant site, with an exclusion radius of one-half mile from the plant.

Cost Estimate

Based on discussions with RES, this issue could be incorporated at no additional cost into the long-term research plan which has not been finalized. A near-term effort would involve an initial expenditure of NRC research funds ($265,000). Depending on the outcome of the research results, additional NRC and industry funds may be needed to develop a solution(s). Because of the small risk, no other costs were estimated.

The industry has a significant economic incentive to repair surface cracks in their steam generators, before they develop into through-wall cracks. As an example, repair of steam generator surface cracks at the Surry plant involved removal by grinding (repair welding was not necessary) estimated by MTEB to cost approximately $1M. At IP-3 where a small through-wall crack developed in one steam generator, the repairs involved grinding and weld repairs. MTEB estimated the costs to IP-3 was approximately $8M. In neither of these cases were the plants required to go into forced outage situations. However, should a plant be placed into a forced outage situation as a result of through-wall cracks in the steam generators, the average replacement power costs of approximately $500,000/day, in addition to the repair costs, would likely result in costs well in excess of $8M.

Other Considerations

A comparison[843] was made of the plants reported by W as having been inspected for indications similar to the IP-3 flaw with plants that have experienced severe steam generator tube degradation histories.[840] The comparison indicated that, in general, the plants inspected were not plants with histories of severe steam generator tube degradations. Subsequent inspections of the replaced Surry Unit 2 steam generators have revealed intermittent cracks up to 1/4 in. deep.[856] The cracks were in the transition region that was part of the original steam generator. The transition cone wall thickness in this area is 3.4 inches and is required by design to be at least 2 inches. Because these indications were in the original part of the transition cone, the affected material was exposed to the same poor secondary water chemistry discussed earlier. The cracking of three Surry 2 steam generator shells occurred at the same joint as the four Indian Point 3 steam generator shells. The inspections of the joints have predominantly been by UT methods from the outside of the shell. As experienced in some of the BWR stainless steel piping inspections for SCC, the UT indications were incorrectly ascribed to geometric configuration. In this regard, IE Information Notice No. 85-65[858] has informed the industry of the events at IP-3 and Surry and the experience with UT versus magnetic particle examinations related to crack detection in the steam generators. Therefore, subsequent ISI testing of the SGS should be more reliable and thereby further reduce the chance of an SGR.

Conclusion

Based on limited operating experience (one steam generator leak in U.S. domestic plants and one steam generator leak in foreign plants) and expert opinion,[859] steam generator outer shells are more likely to leak than to catastrophically rupture. A significant leak in a steam generator outer shell would be expected to result in plant responses comparable to a transient induced by the inadvertent full-opening of the turbine bypass valves. A larger steam generator leak (small rupture) is expected to be bounded by the MSLB with concurrent SGTRs and SBLOCA as evaluated in NUREG-0937.[860] The detailed analyses[860] determined that such an event would not result in a core-melt accident.

To further bound the probability and consequences of this issue, we have ignored the steam generator crack detection experiences and steam generator leak experiences (that essentially have provided defense-in-depth mitigations to severe steam generator ruptures) and assumed a catastrophic SGR probability of 10^{-3}/RY that leads to a LBLOCA (failure of primary piping loop). Based on this scenario as a bounding analysis, the public risk from an SGR was estimated to be 12 man-rem/PWR. Therefore, the risk reduction potential (3.5 man-rem/BWR plant, 12 man-rem/PWR plant) indicates that this issue is of low safety significance to the public.

The quantified values used in this evaluation contain a number of unquantified uncertainties. However, to the extent judged reasonable, the bounding values are believed to be biased in conservative directions. Thus, these estimates are more sensitivity studies than absolute quantifications and, therefore, only represent the potential safety significance of this issue relative to other issues.

We have also considered other concerns raised by MTEB.[857] "The experience at two plants (IP-3 and Surry 2) of the material failure mechanism that was not addressed in the original design (and raised doubt whether GDC 4 is being met) requires a response by the staff. The research effort promised in the future would be too late to address licensing concerns now, especially for operating plants. Active consideration should be given to placing a higher priority on research efforts to enhance our understanding in order to provide a meaningful, timely response." However, MTEB also concluded[857] that this issue only provides a minimal risk to the public health and safety in terms of the contribution to core-melt probability.

Based on (1) the low public risk for this issue, (2) the MTEB expert opinion that steam generator leaks are more likely than SGRs[859] (currently supported by the IP-3 and Garigliano experiences), (3) existing staff recommendations to the industry to implement improved secondary water chemistry programs,[850] (4) the OIE Information Notice[858] that should promote more reliable steam generator inspections, and (5) the industry economic incentive for resolution, this issue has minimal public risk that will be even further reduced by implementation of the above actions.

However, the MTEB concerns related to the need for a better understanding of the materials cracking phenomenon, potential licensing position(s) related to meeting the original licensing design bases, and whether or not the GDC are met, are considered licensing concerns. Therefore, based on the above evaluations, staff actions already taken, and the above discussions, this issue was classified as a Licensing Issue.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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ISSUE 119: PIPING REVIEW COMMITTEE RECOMMENDATIONS

In an August 1983 memorandum,[834] the EDO requested a comprehensive review of NRC requirements in the area of nuclear power plant piping. In response to this request, the NRC Piping Review Committee (PRC) was formed to review and evaluate existing regulatory requirements to: (1) provide recommendations on where and how the NRC should modify requirements; and (2) identify areas requiring further action. The scope of the PRC review covered piping in safety-related systems and high energy lines important to safety in new and operating plants. With respect to postulated pipe breaks, the scope covered all high energy lines.

An NRC steering committee consisting of members from RES, NRR, OIE, and ELD was formed to review and develop a plan for implementing the changes recommended in the PRC report.[611] The steering committee agreed to focus its attention on the recommended research and regulatory changes designated in the PRC report[611] as Category A (high priority) recommendations. The PRC-recommended research and regulatory changes were restructured by the steering committee (combining of research and regulatory recommendations) to form 9 tasks to be addressed by the NRC implementation plan,[835] 5 of which are addressed below. These 5 tasks consist primarily of NRR regulatory actions and some closely-related research efforts. The remaining 4 tasks of the NRC implementation plan related only to research activities and were excluded from this issue.

The five parts of this issue primarily involve revisions to Regulatory Guides and the SRP.[11] No significant change in public safety was expected to result from resolution of this issue; however, resolution of the various tasks was expected to result in less complex and more realistic approaches to piping design and operation in nuclear power plants. The results were expected to yield more efficient regulatory practices, improve plant piping systems design, increase plant reliability, and decrease ORE associated with inspections and repairs. The NRC steering committee agreed that, based on the information provided in NUREG-1061,[611] this work should continue on a schedule consistent with high-priority issues. Therefore, this issue was classified as a Regulatory Impact issue. RES took the lead responsibility for resolution of this issue with assistance from other NRC Offices.[835] The following is an evaluation of the 5 parts of this issue.

ITEM 119.1: PIPING RUPTURE REQUIREMENTS AND DECOUPLING OF SEISMIC AND LOCA LOADS

Description

This task combined two PRC Category A regulatory recommendations with one PRC Category A research recommendation. The designations of the three PRC recommendations were: (1) leak-before-break (A-1); (2) decoupling of seismic and LOCA loads (A-5); and (3) completing research on decoupling (A-4).

One part of the task involved rulemaking changes to GDC-4 in Appendix A of 10 CFR 50 to redefine the need to consider the dynamic effects of pipe breaks. A proposed rule to modify GDC 4 was published[1087] in July 1985 and codified leakbefore-break technology, but was limited only to the primary loop piping of PWRs; the final rule was published[1340] in April 1986. A proposed broad scope rule dealing with all high energy piping in LWRs was published[1341] in July 1986; the final rule was published[1342] in October 1987. With the issuance of these revised rules, revisions to SRP[11] Sections 3.6.1 and 3.6.2 were needed to eliminate the postulation of arbitrary intermediate breaks. The second part of this task involved relaxation of the requirement to consider LOCA and seismic loads simultaneously. A revision to SRP[11] Section 3.9.3 was to be pursued to decouple seismic and pipe rupture loads in the mechanical design of components and their supports.

The existing GDC-4 requirement and SRP[11] Section 3.6.2 pertaining to postulated double-ended guillotine breaks (DEGB) of the largest pipes and postulated arbitrary intermediate pipe breaks needed to be changed to include more realistic criteria and to allow consideration and acceptance of validated analysis methods. The requirements of GDC-4 led to a situation where protective devices were added to forestall events that are extremely unlikely. These protective devices that were designed for the extremely unlikely events could, however, reduce safety and increase worker radiation exposure under normal operations and design basis events.

SRP[11] Section 3.9.3 requires that piping systems and associated components be designed for the combined effects of an SSE and a LOCA. The evolution of seismic design requirements and the calculations of pipe rupture loads have significantly increased the resultant loads obtained by combining these effects. However, field evaluations of piping at conventional power plants and petrochemical facilities indicated that ruptures in piping of the type found in nuclear power plants do not occur during severe earthquakes. Therefore, the staff believed that relaxation of these requirements at all LWRs would not affect plant or public safety.

Conclusion

This task was classified as a Regulatory Impact issue that resulted in revisions[1343,1344] to SRP[11] Sections 3.6.1 and 3.6.2. In addition, Generic Letter No. 87-11[1345] was issued to licensees on the relaxation in arbitrary intermediate pipe rupture requirements (SRP Section 3.6.2). In 1986, the staff terminated[1345] all work on a proposed revision to SRP[11] Section 3.9.3. Thus, this issue was resolved.

ITEM 119.2: PIPING DAMPING VALUES

Description

Historical Background

This task combined PRC regulatory recommendation A-2 (modify seismic damping values used in seismic designs) and PRC research recommendation B-3 (complete research on damping tests). It constituted a two-level approach that could affect all LWRs: a short-term plan and a long-term plan. The short-term action called for a revision to Regulatory Guide 1.84[1347] as the vehicle for NRC endorsement of ASME Code Case N-411. The long-term action called for revisions to Regulatory Guide 1.61[1348] and SRP[11] Section 3.9.2 to incorporate, not only ASME Code Case N-411, but also new positions on pipe damping for high-frequency loads and for time-history analyses.

The short-term endorsement of the ASME Code Case N-411 was to be restricted to seismic response analysis, but not time-history analysis. The long-term action was to result in extensive changes to SRP[11] Section 3.9.2 and Regulatory Guide 1.61[1348] to provide more comprehensive guidance on pipe damping for both seismic and BWR hydrodynamic loadings. Criteria for other non-seismic dynamic loads could also be addressed in the SRP[11] Section 3.9.2 revision.

In general, dynamic piping response could be more accurately predicted if use was made of higher piping damping values than those identified in the existing regulatory guide. The use of higher damping values would result in nuclear plant piping systems having significantly less snubbers and supports and an overall better balance of design, considering all piping loads. A decrease in the number of snubbers and supports could allow better inspection of equipment and components at significantly reduced ORE.

Conclusion

The staff originally planned to take the lead in developing improved pipe damping values and classified the task as a Regulatory Impact issue. However, with the cooperative effort of EPRI, ASME, and the NRC in pursuing the concern, the staff concluded that the most effective approach to the use of more realistic damping values for dynamic piping analysis was through ASME III, Appendix N. When this appendix is completed, the staff will make a decision on its endorsement. As a result, the issue was dropped from further pursuit.[1336]

ITEM 119.3: DECOUPLING THE OBE FROM THE SSE

Description

This task corresponds to PRC regulatory recommendation A-3 (decouple OBE from SSE). 10 CFR 100, Appendix A, Section V(a)(2), stipulates that "(t)he maximum vibratory ground acceleration of the OBE shall be at least one-half the maximum vibratory ground acceleration of the SSE." Therefore, the existing requirement implies the coupling of the two earthquake design levels: SSE and OBE. In developing the existing regulations, it was assumed that the SSE would control the design in nearly all aspects and that the OBE would serve as a separate check of those systems where continued operation was desired at a lower level of ground motion. However, in practice, the assumed load factors, damping, stress levels, and service limits have caused the OBE, rather than the SSE, to control the design for many systems including concrete and steel structures and nuclear piping. In addition, seismic design for OBE accounts for certain safety-related factors such as fatigue and seismic anchor movement that are not considered in the design for the SSE.

Decoupling of the OBE from the SSE or modification of the associated load factors, etc., would impact the design of new plants and would extend well beyond piping considerations. The actions required to resolve this task include: (1) rulemaking to amend and revise Appendix A to 10 CFR 100 to permit decoupling of the OBE and SSE and to incorporate the use of probabilistic methodology in earthquake design; (2) revising and developing Regulatory Guides; (3) updating pertinent sections of the SRP[11]; and (4) advising various industry code committees to revise appropriate codes and guides to reflect changes in the regulations.

A complete listing of the Regulatory Guides and SRP Sections that may be affected by this task were to be identified during the review phase of this task and the related tasks contained in the NRC implementation plan[835] which is of much broader scope.

There is no technical basis for coupling the OBE with the SSE. Designing the piping systems to the SSE is the primary means of ensuring safety. Additional margin is provided by specifying the OBE and thus the level at which inspections will be required before continued operation would be permitted. The more realistic approach of using specific probabilities (return periods) for OBE and the decoupling of the OBE levels and frequencies from those of the SSE will allow assurance of public safety to be placed on a more rational basis.

Conclusion

This item is a Regulatory Impact issue that was integrated[775] by RES into a revision to 10 CFR 100, Appendix A.

ITEM 119.4: BWR PIPING MATERIALS

Description

This task corresponds to PRC regulatory recommendation A-4 to replace regular grade 316SS and 304SS materials in BWR recirculation piping with an alloy resistant to IGSCC. The NRR action related to this task involved preparation of Revision 2 to NUREG-0313[750] and evaluation of each licensee's actions in compliance with this revision.

IGSCC in BWR piping has occurred in a range of piping sizes over the last 25 years and has resulted in major reactor outages. The risk studies reported[611] indicate that pipe failures, even assuming the higher rates due to IGSCC, would not be a major contributor to core-melt and public risk. However, use of materials more resistant to IGSCC should significantly reduce levels of ISI and reactor outage times. Therefore, plant outages and recurring ORE could be significantly reduced by resolution of this task.

Conclusion

This item is a Regulatory Impact issue that required[1506] updating of Regulatory Guide 1.44[1507] by RES to reflect the staff's findings in NUREG-0313,[750] Revision 2, as recommended[925] by NRR. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM 119.5: LEAK DETECTION REQUIREMENTS

Description

This task corresponds to PRC regulatory recommendation A-6 (leak detection requirements). To accomplish this task, additional data are necessary to further validate and improve existing leak-rate prediction analyses. Of particular interest would be investigation and improvement of local leak detection systems such as acoustic emission monitors or moisture-sensitive tapes. These latter techniques may be important for establishing the validity of leak-before-break at specific locations in certain piping systems. The task requires a combination of two approaches: (1) the surveying of operating plants to determine the adequacy of existing leak detection systems; and (2) completion of the research recommended by the PRC and applying the results of the research to regulatory requirements. Subsequent to the completion of key elements of the research effort, the regulatory actions may include the following:

(1) Identify required TS changes such as: (a) unidentified leakage limits for BWRs and PWRs in the context of locating and detecting leakage from cracks with margin; (b) adequacy of surveillance requirements and calibration of systems; (c) alarms; (d) TS consistency; (e) new systems or different detection system combinations; and (f) forward-fit and backfit considerations.

(2) Revise SRP11 Section 5.2.5 and Regulatory Guide 1.45.603

(3) Issue NUREG-0313,750 Revision 2.

It was believed that resolution of this task could affect all LWRs to varying degrees.

No direct safety significance could be attributed to this task. However, knowledge of the leak rates associated with various postulated through-wall crack lengths and confidence in the ability to detect leakage in a timely manner are important elements of the leak-before-break concept that eliminates the postulated DEGB.

Conclusion

This item is a Regulatory Impact issue. As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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ISSUE 127: MAINTENANCE AND TESTING OF MANUAL VALVES IN SAFETY-RELATED SYSTEMS

Description

Historical Background

This issue was identified in the U.S. Nuclear Regulatory Commission (NRC) incident investigation team (IIT) report on the loss of integrated control system (ICS) power event at Rancho Seco Nuclear Generating Station (Rancho Seco) on December 26, 1985 (NUREG-1195, “Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985,” issued February 1986).[1006] Following the event, it was requested that the adequacy of the maintenance program for manual valves be prioritized as a generic issue.[1007] In addition, an information notice[1008] was drafted by the staff and was later issued as IE Information Notice 86-61, “Failure of Auxiliary Feedwater Manual Isolation Valve,”[1010] on July 28, 1986.

Safety Significance

In the Rancho Seco event, when power was lost to the ICS, the plant responded as designed—the auxiliary feedwater (AFW) ICS flow control valves as well as other valves went to the 50-percent open position. However, AFW flow was excessive and an unsuccessful attempt was made to manually close the flow control valve to the “A” once-through steam generator. The operator then attempted to close the manual isolation valve and failed to do so because the valve was frozen in the open position and could not be moved even when a valve wrench was used. Consequently, the inability to reduce AFW flow resulted in an overcooling event. The IIT found that the failure of the AFW manual isolation valve was the result of a lack of preventive maintenance (including lubrication) on this valve during the entire operational life of the plant (about 10 to 12 years).

The manual isolation valve is a locked-open valve located in the AFW discharge header to the “A” once-through steam generator. During the IIT investigation, a Sacramento Municipal Utility District (SMUD) representative stated that the entire AFW system, which would include this manual isolation valve, is safety-related. However, from other discussions with SMUD personnel, it appeared that this valve was only intended to be used to isolate the AFW (ICS) flow control valve for maintenance. The valve is categorized as an American Society of Mechanical Engineers (ASME) Category E valve (i.e., it is normally locked open to fulfill its function). The 1974 edition of the ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, requires no regular testing of Category E valves. The position of the valves is merely recorded to verify that each valve is locked or sealed in its correct position. The current edition of the ASME Code, Section XI, no longer includes a Category E for valves.

Following the incident, it was found that licensees did not have a regular maintenance program that applies to every manual valve. The NRC did not have a requirement for maintenance and testing of convenience valves such as the locked-open manual valve involved in the Rancho Seco incident. The ASME Code, Section XI, specifies inservice inspection, testing, repair, and replacement of valves that are components in systems classified as ASME Classes 1, 2, and 3 and are required to perform a specific function in shutting down a reactor to a cold shutdown condition or in mitigating the consequences of an accident. Manual valves in safety-related systems that are classified as Quality Group A, B, or C in conformance with Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,"[233] are constructed to ASME Code, Section III, Classes 1, 2, or 3 or to earlier codes and standards, as appropriate. These manual valves may be fill, vent, drain, or convenience valves and are constructed to the same code class as the system, or part of a system, of which they are a part. Such valves are not included in the inservice testing (IST) program for valves that are in conformance with the ASME Code, Section XI, as noted above because they are not required to change position to perform a safety function. In the event that a manual valve is required to change position to perform a safety function, it is included in the ASME Code, Section XI, IST program and classified as a safety-related valve.

At the time, the NRC requirements for valve testing were contained in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55(a)(g,) which incorporates the ASME Code, Section XI. Therefore, regulatory requirements for valve testing extend only to valves that are within the IST program. The quality group (safety class) and construction code of each valve are verified, and the valve category is also verified for conformance with the ASME Code, Section XI, Subsection IWV-2000. In addition, the Office of Nuclear Reactor Regulation staff performed a completeness review to assure that all appropriate valves that are within the scope of the ASME Code, Section XI, were included in the IST program. The licensees are responsible for performing the testing, repair, and maintenance of the valves that are within their IST and maintenance programs.

Possible Solutions

The two possible solutions are (1) to develop or revise regulatory requirements relating to the inspection, testing, and maintenance of those fill, vent, drain, and convenience valves in safety-related systems that do not change position for the systems to perform their safety function, or (2) to identify this as an item for which the NRC has concern, notify the licensees by an information notice, and let them determine the maintenance practices they wish to implement.

Priority Determination

In December 1987, the staff assigned a LOW priority ranking to this issue because of the minimal estimated reduction in public risk resulting from the resolution of this issue. This section presents the NRC staff analysis for prioritizing this issue, which was published in 1995. This analysis, which includes frequency, consequence, and cost estimates and a value/impact assessment, has not been updated in the 2011 revision of this issue.

Frequency/Consequence Estimate

To determine the reduction in core-melt frequency that could result from improving the maintenance of manual valves, the Arkansas Nuclear One, Unit 1, Interim Reliability Evaluation Program (IREP) analysis was used.[366] This plant risk study provides a very detailed list of the cutsets and component failures that could result in system unavailability. After a thorough review, no manual valve faults were found for which the mode of failure was the inability to close the valve.

In retrospect, the absence of any identified failure modes concerning the inability to close a manual valve is not surprising; manual valves are, for the most part, installed to permit the isolation of other components (i.e., pumps and motor-operated valves) to permit testing or maintenance without the necessity of shutting the plant down. Hence, they are generally not used for normal or planned emergency operations to control fluid flow. The principal modes of failure associated with manual valves that are identified in risk analyses are either the blockage of a valve or the failure to restore a valve to the open position after it was closed for test or maintenance. In general, most manual valves of the category being considered in this issue are locked in the open position to minimize the chances for inadvertent closure.

Another reason for not finding the failure mode for manual valves in the IREP study[366] is that credit was not given for unplanned recovery actions. Planned operations, as used in this report, include both normal and emergency operations that are directed by procedures. Hence, valve use as was attempted at Rancho Seco would be considered an unplanned recovery event.

Last, the expected frequency of any identified cutsets in which the failure mode included the failure to close a manual valve may have been less than the selected cutoff or truncation value. Considering the failure combinations necessary to involve a manual valve, such may be the case.

It should not be concluded that there is no contribution to core melt and risk by failures that prevent the closure of manual valves (as was the case in the Rancho Seco event) because of their absence from available risk studies or probabilistic risk assessments (PRAs).

As is evident from the Rancho Seco event, the inability to close a manual isolation valve contributed in part to an overcooling event. However, it is probably justifiable to conclude that the inability to close a manual valve contributes only a small amount (i.e., less than 10^{-6}) to core melt and hence to risk. Due to the lack of any identifiable failure or fault combinations in the PRAs, there is no practical basis on which to quantify in this limited analysis the contribution to core melt and risk resulting from these valve failures.

Cost Estimate

Industry Cost: Approximately 100 manual isolation valves of the ASME Class of the AFW manual isolation valves were identified by SMUD that did not receive periodic preventive maintenance. One valve manufacturer recommends lubrication checks at 6-month intervals and actuations (if only partial) on a monthly basis. It is estimated that 4 man-hours will be expended annually per valve performing preventive maintenance and actuation. Assuming that 100 valves are involved, 400 man-hours will be expended each year at each reactor maintaining this class of manual valves. At $35/hour for maintenance personnel,[1009] the direct maintenance cost amounts to $14,000 per reactor-year (RY). In addition, assuming that 20 hours/RY of additional supervisory time at $45/hour will be directed toward added valve maintenance results in $900 of increased costs. Further, assuming an added $100 for additional administrative costs, the total cost for added valve maintenance will be $15,000/RY. Assuming a 30-year plant life and a 5-percent discount rate, the lifetime plant costs associated with the added maintenance of manual valves would be approximately $230,000.

NRC Cost: The NRC cost is estimated to be similar to that incurred in processing a multiplant action per NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980[98]: $6,000).[1009]

Value/Impact Assessment

Due to the inability to ascertain the expected reduction in public risk, the staff did not calculate a value/impact score; however, the risk from this issue was judged to be very low.

Other Considerations

Due to the low costs associated with maintaining the manual isolation valves, it would appear to be cost effective for plant operators to maintain them as a good practice without a regulatory requirement. The power replacement cost for one day of plant outage that may result from the inability to isolate would pay the plant life costs for isolation valve maintenance. In view of this cost-saving potential, the release of the information notice may resolve this issue.

Conclusion

The NRC staff conducted a review of this issue in 2010 to determine whether any new information would necessitate reassessment of the original prioritization evaluation.[1964] The staff determined that the existing regulations and guidance adequately address this issue and the operating experience has not indicated a change in the significance of this issue. The following discussion demonstrates the application of the NRC regulatory framework to this issue.

As published in 1991, paragraphs (a) and (b) of the Maintenance Rule (10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” state the following:

> (a)(1) Each holder of an operating license for a nuclear power plant under this part and each holder of a combined license under part 52 of this chapter after the Commission makes the finding under § 52.103(g) of this chapter, shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in § 50.82(a)(1) or 52.110(a)(1) of this chapter, as applicable, this section shall only apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions….
>
> (b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:
>
> (1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor

> and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in Sec. 50.34(a)(1), Sec. 50.67(b)(2), or Sec. 100.11 of this chapter, as applicable.
>
> (2) Nonsafety related structures, systems, or components:
>
> (i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or
>
> (ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or
>
> (iii) Whose failure could cause a reactor scram or actuation of a safety-related system.

The regulations at 10 CFR 50.65(b)(2)(i) and (b)(2)(ii) address the event presented in this generic issue and, as demonstrated above with applicable operating experience, has addressed similar subsequent events. Moreover, the Standard Review Plan[11] (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition") was revised in 2007 to include Section 17.6, "Maintenance Rule," which outlines the criteria for evaluating licensee applications for the scope, monitoring, evaluation, and risk assessment and management of implementing 10 CFR 50.65, including Section III, 1.B, which outlines the criteria for including nonsafety-related structures, systems, and components (SSCs) in accordance with 10 CFR 50.65(b)(2). Criterion iii of this section applies directly to this generic issue, stating that the description of the maintenance rule scoping process should address the following:

> SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related functions in accordance with 50.65(b)(2)(ii). The applicant should describe how the process considers system interdependencies, including failure modes and effects of nonsafety-related SSCs (*e.g.*, support systems) that could directly affect safety-related functions.

Based on the review of the NRC's regulations and guidance related to this issue, the staff concluded that existing regulations and guidance adequately address this issue. Therefore, the staff changed the status of Generic Issue 127 and DROPPED this issue from further pursuit.

References

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ISSUE 155: GENERIC CONCERNS ARISING FROM TMI-2 CLEANUP

The TMI-2 Safety Advisory Board was established to provide the licensee, General Public Utilities Nuclear Corporation, with a qualified, independent appraisal of the cleanup of TMI-2, with particular emphasis on the assurance of public and worker health and safety. As a result of this appraisal, seven recommendations[1362] were forwarded to the NRC for evaluation. These recommendations were treated as separate generic issues as outlined below.

ISSUE 155.1: MORE REALISTIC SOURCE TERM ASSUMPTIONS

Description

During the TMI-2 accident, fission products did not behave as predicted with the analytical methods and assumptions used in the licensing process at that time and delineated in Regulatory Guides 1.3[213] and 1.4[214] and TID-14844.[73] The earliest expert predictions were that major core damage had occurred. However, the NRC and the licensee believed that core damage was minimal and calculations were redone to confirm this view. Approximately 50% of the core was in a molten state, but there is evidence that only about 55% of the highly volatile fission products and noble gases were released from the reactor vessel with a major portion retained in the reactor building. There is also evidence that less than 5% of the medium and low volatile fission products were released from the reactor vessel.[1362] These observations were based on research conducted since the TMI-2 accident.

It is now generally accepted that the chemical conditions in the reactor vessel were "reducing" in nature as opposed to "oxidizing." The elemental iodine was driven (or converted) to the iodide ion which very readily combined with available metallic ions. The water-soluble character of these chemical forms prevented a major release of iodine to the atmosphere of the containment or auxiliary buildings and only a few Curies were released to the environment. Throughout the TMI-2 accident sequence, the chemical state was maintained such that the water-soluble character was preserved.

With the completion of a large number of PRAs since the TMI-2 event, the Advisory Board believed that it should be possible to list accident sequences with chemical conditions similar to TMI-2. Such a listing could provide a guide as to which accidents might be regarded as hazardous, or less hazardous, relative to the possible escape of iodine and could be useful in the future design of safety features. Since some of the assumptions used for source term considerations at TMI-2 were flawed in this respect, the Board recommended that the source term be restated using current scientific knowledge.[1362]

Conclusion

At the time this issue was evaluated in February 1992, comprehensive revisions to 10 CFR Parts 50 and 100 were being pursued by the staff to reflect a better understanding of accident source terms and severe accident insights, as well as evaluate the impact of these phenomena on plant engineered safety features. A replacement for TID-14844[73] was being formulated, based on previous severe accident research findings, to reflect the existing understanding of fission product release timing, iodine chemistry, and source term magnitude and composition. Thus, a solution to this issue had been identified and the issue was considered nearly-resolved.

In resolving the issue, the staff issued NUREG-1465[1465] which provided more realistic estimates of the fission product source term release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe core-melt accident. Thus, the issue was RESOLVED with new requirements for future plants.[1530] In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period would not affect the resolution.

ISSUE 155.2: ESTABLISH LICENSING REQUIREMENTS FOR NON-OPERATING FACILITIES

Description

At the time the TMI-2 event occurred, 10 CFR 50 contained regulations primarily for the design, construction, and operation of nuclear facilities but did not provide adequate guidance for the post-accident condition. Much was learned while the unit was being defueled and prepared for the post-defueling, monitored storage phase. The decommissioning rule[1364] issued in 1988 addressed the safe removal of nuclear facilities from service and the reduction of residual radioactivity to a level that permits release of the property for unrestricted use and termination of the operating license. The options for compliance with this rule are described in NUREG-0586[173] and include DECON, SAFSTOR, and ENTOMB. Decommissioning activities do not include the removal and disposal of spent fuel; these are considered to be operational activities.

Once a reactor is permanently shut down and defueled, it enters a storage phase until the licensee begins implementation of a decommissioning plan approved by the NRC. During the storage phase, requirements for security plans, operator licensing, emergency planning, etc., that were in effect while the plant was operational, may become unnecessary and burdensome to the licensee. Once all nuclear fuel is removed from the reactor site, the risk of an extraordinary accident, as defined in 10 CFR 50.54(w) and 10 CFR 140.11, is essentially eliminated. The Board recommended that regulatory guidance be developed for use by non-operating and defueled facilities during the storage phase prior to decommissioning.[1362]

Conclusion

This issue addressed changes in existing regulatory guidance that could significantly reduce licensee costs without any substantial change in public risk. Thus, it was classified as a Regulatory Impact issue. Staff stated in the Supplement to NUREG-0933 published in 1995 that revisions to 10 CFR 50.54(w) and 10 CFR 140.11(a)(4) might be necessary to address insurance coverage for non-operating and defueled facilities during the storage phase prior to decommissioning.[1363]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ISSUE 155.3: IMPROVE DESIGN REQUIREMENTS FOR NUCLEAR FACILITIES

Description

The Board recommended[1362] that the NRC undertake an effort to evaluate lessons learned at TMI-2 and incorporate them into the design of future nuclear plants. The recommendations suggested by the Board focused on recovery from a severe accident and were as follows:

(1) Prohibit the use of cinder blocks inside the reactor building (because they absorb so much contamination and become a radiological hazard) or designing the facility to be "robot friendly."

(2) Utilize higher range radiation instrumentation in order to monitor the environment inside the reactor building during a severe reactor accident.

(3) Based on design criteria and clear evidence that the TMI-2 containment building was not challenged, a reduction in criteria might be prudent based upon actual accident conditions. The NRC had reviewed in some detail the capability of reactor containment structures to withstand accident environments, including significant pressure increases; a review of these studies might be helpful and may lead to a reduction in design criteria. A similar effort for reactor vessels has not been undertaken and should be, considering the condition of the lower head of the TMI-2 reactor vessel with the severity of the accident.

(4) TMI-2 has also demonstrated the need to provide access to the underside of a reactor vessel for remote inspections to determine the extent of possible damage in the aftermath of a severe reactor accident. The 52 instrument penetrations in the lower head of the TMI-2 reactor vessel have been a concern since the discovery of once-molten material on the lower head of the reactor vessel and thus lower head integrity has been a major concern during the recovery efforts. For future reactor vessel design, it was recommended that in-core instrumentation penetrate the head instead of the bottom.

Priority Determination

The four concerns outlined in this issue were evaluated separately below:

(1) In accordance with 10 CFR 50, Appendix I, nuclear power plants are required to keep occupational risk exposure (ORE) as low as is reasonably achievable (ALARA). Cinder blocks constitute one of the materials that are used inside the reactor building of some operating plants as local shielding to meet this ALARA criterion. Prohibiting the use of cinder blocks inside the reactor building would have no impact on public risk in the event of a severe accident. The use of other shielding materials that do not absorb as much contamination has the potential for decreasing the decontamination time (and ORE) following a severe accident.

Designing future nuclear plants to be robot-friendly will require spatial considerations for the mobility of robots that could drastically increase design, engineering, and construction costs. However, as is the case above, the use of robots would have no impact on public risk in the event of a severe accident; only occupational risk would be affected.

From NUREG/CR-2800,[64] the occupational dose from cleanup, repair, and refurbishment following a severe accident was estimated to be 19,860 man-rem. Even assuming that 50% of this dose can be reduced with either the elimination of cinder blocks or the use of a robot for cleanup and assuming a core-melt frequency of 10^{-5}/RY and an average remaining reactor life of 28 years, the potential dose reduction is approximately 3 man-rem/reactor. Thus, this concern had negligible risk reduction potential and consideration of costs would only lower its priority ranking.

(2) The recommendation to utilize higher range radiation instrumentation in order to monitor the environment inside the reactor building during a severe accident was addressed by TMI Action Plan Item II.F.1. This item was clarified in NUREG-0737[98] and required implementation at all plants. Thus, this concern was previously addressed by the staff.

(3) For future plants, the Commission's Severe Accident Policy Statement established the criteria and procedural steps under which new designs for nuclear power plants could be acceptable for meeting severe accident concerns. Rather than a reduction of criteria, it is expected that future plants would have to achieve a higher standard of severe accident safety performance, including clarification of containment performance. The staff's plan of action in this area was presented to the Commission in SECY-92-292.[1427] Operating plants were assessed under the Containment Performance Improvement Program (see Issue 157).

The mode of vessel failure, including investigation of the TMI-2 vessel, was being pursued by the staff as part of its severe accident research program.[1382] The results of this research was expected to determine whether changes to future vessel design would be warranted. Thus, this concern was being addressed by the staff.

(4) The relocation of in-core instrumentation was expected to be addressed by NSSS vendors in the design of future plants which was subject to review and approval by the staff. For example, the bottom-mounted instrumentation penetrations were eliminated in the Westinghouse AP600 design to reduce building volume and costs significantly. Thus, this concern was being addressed by the staff.

Conclusion

Of the four recommendations contained in this issue, two were addressed in other ongoing programs and one had been previously addressed by the staff. The remaining recommendation had negligible risk reduction potential and, therefore, was not considered to be safety-significant. Thus, this issue was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 155.4: IMPROVE CRITICALITY CALCULATIONS

Description

The Board believed that doubts still remained as to whether the TMI-2 core became critical, or was very close to critical, during the TMI-2 accident and recommended that the NRC establish guidelines that deal with criticality following a severe reactor accident.[1362] These guidelines should take into account abnormal geometries and possible core conditions that could result from the accident. The Board believed that the accident scenario developed by the TMI-2 licensee was sufficiently detailed that a series of geometric configurations could be simulated for criticality calculations. Variables that could be estimated reasonably well included the presence of water, oxidation of cladding, melting and movement of fuel, melting of poison rods, and movement of poison.

Conclusion

The safety concern was addressed by DSR/RES in SARP Task 4.3: Investigate the Possibility and Consequences of Recriticality in Degraded BWR Cores.[1382] The staff's study was documented in NUREG/CR-5653[1379] in which it was concluded that there was the potential for recriticality in BWRs, if core reflood occurs after control blade melting has begun but prior to significant fuel rod melting. However, a recriticality event would most likely not generate a pressure pulse significant enough to fail the vessel. Two strategies were identified that would aid in regaining control of the reactor and terminate the recriticality event before containment failure pressures are reached: (1) initiation of boron injection at or before the time of core reflood, if the potential for control blade melting exists; and (2) initiation of RHR suppression pool cooling to remove the heat load generated by the recriticality event and extend the time available for boration.

The issue was not considered to be a major concern for PWRs because of their design that includes a safety injection system for supplying borated water to the core. Furthermore, it was concluded in NUREG/CR-5856[1417] that, during a severe accident, an unmoderated recriticality of the molten, consolidated portion of a degrading core cannot occur at U_{235} enrichments characteristic of a PWR. Based on the staff's efforts in addressing the safety concerns in the SARP, this issue was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 155.5: MORE REALISTIC SEVERE REACTOR ACCIDENT SCENARIO

Description

The TMI-2 event was a severe accident in which approximately 50% of the core was in a molten state at some point during the accident. Approximately 20 tons of the once-molten debris poured through the core support structure into the water-filled lower plenum and onto the lower head of the reactor vessel. Most codes in use at that time would have predicted a failure of the lower head under these conditions. The severity of the accident showed that the reactor vessel was more difficult to fail than was anticipated.

The Board recommended that in-vessel core-melt progression for severe accidents be studied further by the NRC and that the results be incorporated into existing codes and standards. The Board believed that codes should have the capability to reproduce the TMI-2 accident with reasonable accuracy before they can be accepted as predictive tools.[1362]

Conclusion

At the time this issue was evaluated in June 1992, the safety concern was being addressed by DSR/RES in SARP Issue L2: In-Vessel Core Melt Progression and Hydrogen Generation.[1382] In considering core-melt progression, the staff was expected to treat BWRs and PWRs separately because of their different fuel assembly, control element, and lower plenum structures. Concerns common to both BWRs and PWRs are: (1) the integrity of core structures; (2) the mode of core material relocation; (3) hydrogen generation; (4) the mode of bottom head failure; and (5) the effects of water injection. The answers to the above concerns will be different because of the physical differences of BWRs and PWRs. TMI-2 data and the results of new experiments and model development were to be examined by the staff in its research. Based on the staff's efforts on SARP Issue L2, Issue 155.5 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 155.6: IMPROVE DECONTAMINATION REGULATIONS

Description

The Board believed that the decontamination techniques used throughout the nuclear industry for small activities were not applicable to large-scale activities and recommended that the NRC use the experience gained from the TMI-2 accident to prepare guidelines for decontamination and decommissioning of nuclear plants.[1362]

Conclusion

Traditionally, the NRC has not developed or approved decontamination techniques. Due to the many ways in which decontamination can be accomplished and the rapidly evolving technology in this area, it is not practical or beneficial for the NRC to establish guidelines for decontamination techniques. Rather, the NRC has focused on the development of criteria which set standards for exposure of workers and the public (e.g., 10 CFR 20), the levels of allowable residual contamination, and the handling and disposal of the radioactive waste generated. Efforts at establishing residual contamination criteria applicable to decommissioning were in progress as described below.

In June 1991, the Commission deferred[1412] implementation of the Below Regulatory Concern (BRC) policy but reaffirmed its intentions to carry out its responsibilities to address issues related to waste disposal, consumer products, recycling of materials, and decontamination and decommissioning, as necessary, on a case-by-case basis in the manner in which these issues were considered, prior to the development of the BRC policy statement. In this regard, the staff was directed to continue its accelerated efforts in completing the technical basis for rulemaking on residual contamination criteria.

In accordance with SECY-92-045,[1413] the staff proceeded with an enhanced participative rulemaking process to develop radiological criteria for decommissioning; this effort was tracked in the NRC Regulatory Agenda (NUREG-0936). Based on the above considerations, Issue 155.6 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 155.7: IMPROVE DECOMMISSIONING REGULATIONS

Description

The Board raised concerns over the requirements for the disposal of highly contaminated components from a nuclear plant during decommissioning and recommended that regulations be developed.[1362]

Conclusion

The TMI-2 experience was considered by the staff in the development of the decommissioning rule[1364] in 1988. Industry options for complying with this rule are described in NUREG-0586[173] and include DECON, SAFSTOR, and ENTOMB. As part of its resolution of Issue B-64, "Decommissioning of Reactors," the staff is currently developing an SRP[11] Section for use in its review of licensee decommissioning plans. Concurrent with this effort is the development of two Regulatory Guides: DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors"; and DG-1006, "Records Important for Decommissioning of Nuclear Reactors." Thus, Issue 155.7 was DROPPED from further consideration as a new and separate issue. The related concern of decommissioning prematurely shutdown plants was addressed in Issue 155.2. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

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1362. Memorandum for E. Beckjord from F. Gillespie, “Generic Concerns Arising from TMI-2 Cleanup,” February 21, 1991. [9103010101]

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1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

ISSUE 167: HYDROGEN STORAGE FACILITY SEPARATION

Description

Historical Background

Issue 106 was resolved with the issuance of Generic Letter 93-06, "Research Results on Generic Safety Issue 106, 'Piping and the Use of Highly Combustible Gases in Vital Areas,'" dated October 25, 1993,[1547] which included evaluation of the risk from (1) the storage and distribution of hydrogen (H_2) for the volume control tank in pressurized-water reactors (PWRs) and the main electric generator in boiling-water reactors (BWRs) and PWRs, (2) other sources of H_2 such as battery rooms, the waste gas system in PWRs, and the offgas system in BWRs, and (3) small, portable bottles of combustible gases used in maintenance, testing, and calibration. However, the potential risk from large H_2 storage facilities outside the reactor, auxiliary, and turbine buildings was not addressed. Studies performed during and subsequent to the resolution of Issue 106 raised concerns about the magnitude of the excluded risk.[1534,1535] Thus, in December 1993, Issue 167 was identified[1532] to address this excluded risk.

U.S. Nuclear Regulatory Commission (NRC) Information Notice 89-44, "Hydrogen Storage on the Roof of the Control Room,"[1552] was issued in April 1989, and each NRC regional office was expected to determine whether the plants in its region had similar safety-related concerns. The information compiled by these offices was reviewed and issued in the preliminary report, SCIE-EGG-103-89, "Draft Technical Evaluation Report on U.S. Commercial Power Reactor Hydrogen Tank Farms and Their Compliance with Separation Distance Safety Criteria," in March 1990.[1535] The storage of gaseous or liquid H_2 at 119 power plants was then investigated, and possible accident scenarios resulting from a fireball, explosion, or presence of unburned H_2 gas in ventilation air intakes were examined. Explosion was identified as the scenario posing the greatest risk potential. The analysis in SCIE-EGG-103-89[1535] focused on explosion, with all quantification performed relative to this accident only.

Safety Significance

The safety concern was whether or not there is adequate physical separation between H_2 storage facilities and buildings or structures housing systems important to safety at nuclear power plants.

Possible Solutions

Possible solutions included relocation (or placement in pits) of storage facilities, buildings, and equipment and the construction of blast shields, or a combination of these. The resolution for this issue was assumed to be the construction of concrete walls enclosing the H_2 storage facility. This structure would serve as a blast shield in the event of an explosion, essentially eliminating the risk.

Priority Determination

The NRC staff assigned a LOW priority ranking to this issue in 1994. This section presents the NRC staff analysis for prioritizing this issue, which was published in 1995. This analysis, which includes frequency, consequence, and cost estimates and a value/impact assessment, has not been updated in the 2011 revision of this issue.

Hydrogen gas and cryogenic H_2 storage tanks are designed, fabricated, tested, and stamped in accordance with Section VIII, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for unfired pressure vessels. The containers for gaseous H_2 are seamless, single-walled containers. Liquid H_2 is stored in vacuum-jacketed or double-walled vessels. The "Handbook of Chemical Hazard Analysis Procedures" (the Handbook) (published jointly by the Federal Emergency Management Agency, U.S. Department of Transportation, and U.S. Environmental Protection Agency) lists accident rates and percentages of volume released for use in analyzing potential accidents involving H_2 storage containers. For single-walled containers, the accident rate suggested in this Handbook was 10^{-4}/tank-year. For these containers, the Handbook suggested that 90 percent of spills are terminated, while 10 percent are instantaneous total release of contents. Thus, for single-walled containers, the frequency of release of 100 percent of container contents was estimated to be 10^{-5}/tank-year. For double-walled containers, the accident rate suggested was 10^{-6}/tank-year. In this case, the entire container contents are released instantaneously 100 percent of the time. A frequency of 10^{-5}/tank-year for instantaneous release of 100 percent of vessel contents was assumed in this analysis.

The status of the 119 power plants was assessed[1535] with respect to H_2 tank farm separation guidelines in Electric Power Research Institute (EPRI) NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations." Sixteen percent were found not to meet the separation guidance with respect to explosion hazard. For the existing population of 110 plants, this translated into 18 light-water reactors not meeting the EPRI guidelines. (The permanently shut down Trojan PWR was excluded from this population. Thus, this analysis did not address H_2 storage tanks located on top of the control room roof.) However, in NUREG-1364, "Regulatory Analysis for the Resolution of Generic Safety Issue 106: Piping and the Use of Highly Combustible Gases in Vital Areas," issued June 1993,[1545] credit was assumed for an informal survey that showed mitigating factors to be insufficient at only three plants.

Frequency/Consequence Estimate

Of the types of accidents analyzed in risk assessments, H_2 tank farm explosions seem most similar to some accidents classified as external events. Furthermore, because such explosions could cause large pressure forces to be exerted upon building walls or could cause possible impact by missiles, they would appear to be similar to tornadoes among the types of external events. However, unlike what is usually assumed for tornadoes, H_2 tank farm explosions probably would not exert a uniformly high and destructive pressure on all buildings on site at one time. Thus, the consequences from H_2 tank farm explosions were not expected to exceed those from tornadoes.

A review of available probabilistic risk assessments (PRAs) yielded the individual plant examination (IPE) for Oconee Nuclear Station (Oconee), Unit 3,[1533] as most appropriate for this analysis. In this IPE, a fairly detailed assessment of tornado risk was performed, building on that from the earlier Oconee Unit 3 PRA.[889] An added advantage to the selection of Oconee Unit 3 was that it provided a description of the site's 48,000-scf H_2 tank farm, which was deemed to be representative.[1535] This farm consisted of six tanks that, using the suggested values in the Handbook, resulted in an accident frequency of (10^{-5}/tank-year)(6 tanks/tank-farm)(1 tank-farm/reactor) = $6x10^{-5}$/reactor-year (RY) for an H_2 accident that releases 100 percent of the contents of at least one tank.

To put the amount of H_2 involved in perspective for the tank farm at Oconee, one tank contains 8,000 scf of H_2. This was equivalent to 216.8 pounds (lbs) of trinitrotoluene (TNT), using the EPRI guidelines equivalence of 1,000 scf = 27.1 lbs of TNT for gaseous H_2 storage. Six tanks in the same farm contain an equivalent of 1,300.8 lbs of TNT. In the terminology of the EPRI guidelines, both these amounts are considered to be in the "small equivalence" range (less than 4,000 lbs TNT). Terminology and equivalence notwithstanding, hypothesizing the detonation of one tank in a farm raises the question of subsequent damage to and detonation of adjacent tanks. The EPRI guidelines were based on the safety analysis of the failure of single vessels and did not address simultaneous failure of multiple storage vessels. There was factual support for using a basis of only one tank failure. The guidelines cited three events, two from reactor sites where H_2 container explosions did not damage adjacent cylinders. Given release of the contents of one cylinder, it was assumed in this analysis that (1) detonation will occur, (2) possibly all of the tanks were involved, and (3) because the tank farm in question was assumed not to conform to EPRI guidelines, appropriately selected plant damage would ensue with an appropriately assigned conditional probability.

For the purpose of analyzing tornado-generated missiles, Duke Power Company (Duke Power) considered[1533] two categories of tornado events: (1) tornadoes whose winds impact on Oconee Unit 3, and (2) tornadoes passing within 2,000 feet (ft) of Oconee Unit 3. The latter category was subsequently dismissed when analysis showed the probability of a core-melt due to tornado-generated missiles to be 100 to 1,000 times lower than that due to tornado wind loadings; therefore, only the first category was addressed. Duke Power assumed[1533] that a tornado would render unavailable all offsite alternating current (ac) power sources except for one underground path. Tornadoes of intensity F-1 or less (i.e., with wind speeds less than 113 miles per hour (mph)) were assumed not to cause sufficient wind damage to generate a core melt. Oconee had been designed to withstand wind loadings of F-1 tornadoes.

The EPRI guidelines have been checked for responses for walls with static pressure capacities between 1.5 and 4.5 pounds per square inch (psi). Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants,"[42] indicated that a 1.5-psi pressure drop could be expected for a design-basis tornado with wind speed as low as 195 mph (the sum of rotational speed and minimum translational speed). This wind speed lies toward the upper end of the range for an F-3 tornado (158–206 mph). From Appendix B to the EPRI guidelines, for a small yield such as that (216.8 lbs TNT) from one tank at the tank farm in question, the separation distance can be fairly small (about 60 ft), even for a plant with moderate wall ductility (μ = 3) and low-end (1.5-psi) static design pressure. This lends some justification to the assumption made that the appropriate minimum blast force to use in the analysis corresponds to that from an F-3 tornado. Due to lack of knowledge of the exact number of tanks detonated (one to six) and other physical parameters involved, conditional probabilities of 1/3 were assigned to each of the resulting equivalent tornado forces F-3, F-4, and F-5. In other words, the $6x10^{-5}$/RY initiating event frequency derived from the Handbook was considered to be uniformly distributed among the assumed equally likely outcomes F-3, F-4, and F-5. This was believed to be conservative, at least with respect to F-4 and F-5.

The turbine building was assumed[1533] to be susceptible to wall damage from F-2 and stronger tornadoes. Wall damage could fail the 4,160-volt (V) (4-kilovolt (kV)) ac switchgear that powers safety equipment and/or the upper surge tank (UST), the prime suction source for the emergency feedwater pumps. The auxiliary building was assumed to be susceptible to wall damage from F-4 and stronger tornadoes (wind speeds greater than 206 mph), particularly the exterior walls of the west penetration room (WPR) and east penetration room (EPR). Damage to the WPR wall could fail piping and electrical penetrations, including those from the standby shutdown facility (SSF). This could lead to reactor coolant pump seal loss-of-coolant accidents, loss of the SSF backup for reactor coolant pump seal cooling, and loss of feedwater from the SSF. Damage to the EPR wall would cause similar failures, although the likelihood of piping failures there was judged to be about 10 times less due to tornado shielding by the reactor building. Other exterior components, such as the borated water storage tank, are also susceptible to failure from tornadoes. However, they did not appear in the listed cutsets for core damage and were not considered further.

Tornadoes falling within categories F-2 through F-5 resulted in accident sequences leading to various plant damage states. The total core damage frequency (CDF) (the sum of the accident sequence frequencies) was 9.74×10^{-6}/RY. However, after eliminating the plant damage state not resulting in offsite releases, the frequency was calculated to be 8.2×10^{-6}/RY.

There were 17 possible release categories associated with the plant damage state, and each was assigned a conditional probability of release.[1533] When multiplied by the sequence frequency, each of these yielded the sequence frequency per release category. Associated with each category was a whole-body man-rem equivalent dose. The product of each release category frequency and its associated dose yielded a total risk of 9.11 man-rem/RY. These results for a tornado were then modified for an H_2 tank-farm explosion.

Again, unlike tornadoes, H_2 tank-farm explosions would not exert pressure on all site buildings at one time. Thus, multiple building wall failures were not expected as in the tornado accident sequences. To reflect this limitation, the cutsets of the tornado accident sequences were reviewed and it was found that nearly all contained conditional failure of the turbine building wall. Associated with this failure were failures of the UST and/or 4-kV ac switchgear. Failures of the walls of the WPR and/or EPR were contained in fewer of the cutsets of the tornado accident sequences. Therefore, an H_2 tank-farm explosion was assumed to fail only the turbine building wall, resulting in failures of the UST and/or 4-kV ac switchgear; no failure of the WPR or EPR walls was assumed. This eliminated most of the T(F4) and T(F5) sequences. Using the remaining sequences, the Handbook derived an initiating event frequency of 6×10^{-5}/RY, and the 1/3 conditional probability for each of the categories F-3, F-4, and F-5 resulted in a total CDF frequency from H_2 tank explosion of 4×10^{-6}/RY, which was less than that for tornadoes.

Using the same release category conditional probabilities and equivalent doses as for tornadoes, a total frequency over all release categories of 3×10^{-6}/RY (less than the total CDF because not all accident sequences lead to offsite release) and a total risk of 2.9 man-rem/RY were obtained. Thus, the risk was also less than that for tornadoes. Assuming 18 affected plants with an average remaining lifetime of 23 years, the total risk reduction potential was 1,201 man-rem.

Cost Estimate

Industry Cost: It was reported[1535] that the 48,000-scf Oconee tank-farm consisted of six tanks that covered an area 45 ft by 30 ft. It was surrounded by an exclusion fence and was always lighted. The proposed concrete enclosure was assumed to have the same dimensions as the six Oconee tanks combined. From the 1993 *Means Building Construction Cost Data*

(51st edition), the costs were obtained for thick, smooth, gray, architectural precast concrete slabs 10 ft high and 6 inches thick. For a 20-ft length, the cost was \$14.95 per square foot (ft^2) (area) and, for a 30-ft length, the cost was \$14.60/ft^2. To form an enclosure at least 45 ft by 30 ft, two 20-ft sections and four 30-ft sections would be needed. A height of 10 ft was considered to be sufficient to protect the surroundings from horizontal blast effects; however, a thickness comparable to that of site building walls (about 18 inches) was necessary. Thus, the total number of precast concrete wall panels was as follows:

30-ft panels:	(4/perimeter)(3 at 6-in thickness each) =	12
20-ft panels :	(2/perimeter)(3 at 6-in thickness each) =	6
		Total = 18

The enclosure would be 50 ft by 30 ft, yielding a total wall panel area of (2)(50 + 30)ft(10 ft)(3 panels) or 4,800 ft^2. At \$15/ft^2, the cost of this enclosure would be (\$15/ft^2)(4,800 ft^2) or \$72,000.

The Means manual cited above stated that, "[i]f the work is to be subcontracted, add the general contractor's markup, approximately 10%." In addition, the enclosure will have to be anchored in place and penetrated for piping and access. Combined with the general contractor's markup, these factors were assumed to increase the cost of the enclosure by about 50 percent, bringing the total cost to (1.5)(\$72,000)/plant or \$108,000/plant.

Any industry operation and maintenance activities associated with this resolution were assumed to be performed as part of activities already in place, such as standard inspection and reporting procedures. No industry cost was anticipated for operation and maintenance. Thus, the total cost for 18 plants was \$1.94 million (M).

<u>NRC Cost</u>: The resolution was assumed to require \$100,000 for development of a "typical uncomplicated TS [technical specification] change" and an implementation cost of \$11,000/plant (1988 dollars) with a 4-percent inflation rate. For 18 affected plants, the total cost was estimated to be \$341,000.

<u>Total Cost</u>: The total NRC and industry cost associated with the possible solution was estimated to be \$(1.94 + 0.341)M or \$2.28M.

Impact/Value Assessment

Based on a potential public risk reduction of 1,201 man-rem and an estimated cost of \$2.28M, the impact/value ratio was given by the following:

$$R = \frac{\$2.28\text{M}}{1{,}201\text{ man-rem}}$$

$$= \$1{,}898\text{/man-rem}$$

Other Considerations

The following other considerations relate to this issue:

(1) The number of affected plants was determined by identifying those that did not conform to the EPRI separation criteria that were based on an H_2-to-TNT detonation equivalency. Had more stringent criteria been used, the number of affected plants might have been larger. EGG-SSRE-8747, “Technical Report: Improved Estimates of Separate Distances to Prevent Unacceptable Damage to Nuclear Power Plant Structures from Hydrogen Detonation for Gaseous Hydrogen Storage,” issued November 1993,[1534] concluded that “the hydrogen to TNT detonation equivalency used in previous calculations should no longer be used.” The stated reason for this was that “the separation distances results from previous calculations [including those of the EPRI criteria] can be either overconservative or unconservative depending upon the set of hydrogen detonation parameters that are used.” Nevertheless, this analysis was considered to be sufficiently conservative, particularly with respect to the assumption that all tanks are single-walled and assignment of conditional probabilities of 1/3 for F-4 and F-5 resultant forces.

An informal survey of all plants cited in NUREG-1364[1545] showed that, of those plants that did not meet the EPRI criteria for separation distances for safety-related air intakes or structures, mitigation features were insufficient at only three of the plants. Use of 3 affected plants instead of 18 would also result in a low impact/value score.

(2) Assuming a license renewal period of 20 years, with the 18 affected plants operational 75 percent of this time, the additional risk reduction would be (18)(2.9 man-rem/RY)(20)(0.75) or 783 man-rem. Because there would be no increase in cost, the impact/value ratio would be $1,149/man-rem.

Conclusion

Based on the impact/value ratio and the potential reduction in CDF and public risk, this issue was given a LOW priority ranking in 1994. Consideration of a license renewal period of 20 years would not change this ranking.

The NRC staff conducted a review of this issue in 2010 to determine whether any new information would necessitate reassessment of the original prioritization evaluation.[1964] The staff determined that the existing regulations and guidance adequately address this issue, and the operating experience has not indicated a change in the significance of this issue. The following discussion demonstrates the application of the NRC regulatory framework to this issue.

Between the publication of this generic issue in 1995 and the year 2000, most licensees committed to National Fire Protection Association (NFPA) 50A, “Standard for Gaseous Hydrogen Systems at Consumer Sites,” and NFPA 50B, “Standard for Liquefied Hydrogen Systems at Consumer Sites,” as part of their licensing basis.[1969] These codes provided separation distances for gaseous and liquefied hydrogen, providing a basis for the generic issue.

In 2000, with the implementation of the Reactor Oversight Process, the NRC issued Inspection Procedures 71111.05AQ, “Fire Protection Annual/Quarterly,”[1970] and 71111.05T, “Fire Protection (Triennial).”[1971] These inspection procedures have the following objectives:

- Evaluate the adequacy and implementation of the licensees’ fire protection programs.
- Review the procedures to incorporate and implement changes to the respective fire protection programs.

- Determine the adequacy of the licensees' systems for taking corrective action when warranted by quality assurance programs, generic deficiencies, or events.

With respect to this generic issue, these inspection procedures verify that a licensee's fire protection program includes the control of combustible material, including the appropriate storage of bulk flammable gases and liquids like hydrogen. To that end, inspection procedures also verify that the licensee's fire protection program consists of a fire hazard analysis, which includes analyses for postulated hydrogen explosions. The fire protection program also includes the facility's technical specifications, which include the appropriate limiting condition for operations to prevent the postulated fire conditions.

In December 2002, the NRC reported the results of the inspections under Temporary Instruction 2515/146, "Hydrogen Storage Locations," Revision 1, dated April 18, 2002.[1972] The report highlighted findings related to the adequate separation of hydrogen storage facilities from risk-significant tanks or SSCs and from ventilation intakes. The licensees of these plants committed to taking appropriate corrective actions.

With respect to enforcement, in December 2008, inspectors identified a Severity Level IV noncited violation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, "Changes, Tests and Experiments," for the licensee's failure to perform a safety evaluation associated with installation of a bulk hydrogen storage facility located directly above buried circulating water system return lines.[1973]

Based on the review of the NRC's regulations and guidance related to this issue, the staff concluded that existing regulations and guidance adequately address this issue. Therefore, the staff changed the status of Generic Issue 167 and DROPPED this issue from further pursuit.

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ISSUE 181: FIRE PROTECTION

Description

In February 1993, The U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) completed a reassessment of the reactor fire protection review and inspection programs, in response to programmatic concerns raised during the review of Thermo-Lag fire barriers, and prepared a report.[1626] A fire protection task action plan (FP-TAP) was then prepared to implement the recommendations that resulted from this reassessment. The FP-TAP includes a wide range of technical and programmatic fire protection issues, including recommendations for action (Part I), recommendations for further study (Part II), confirmation issues (Part III), and lessons learned (Part IV). Staff actions to address the recommendations were submitted to the Commission in SECY-93-143, “NRC Staff Actions To Address the Recommendations in the Report on the Reassessment of the NRC Fire Protection Program,” dated May 21, 1993,[1627] and progress reports[1628,1629] were issued. This issue was identified in an NRR memorandum[1601] to the Office of Nuclear Regulatory Research in February 1996.

Each operating reactor has an NRC-approved fire protection plan that, if properly implemented and maintained, satisfies Title 10 of the Code of Federal Regulations (10 CFR) 50.48, “Fire Protection,” and General Design Criterion 3, “Fire Protection,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.” The staff’s focus was on developing the framework for the future direction of the NRC fire protection program, with emphasis on a fire protection functional inspection (FPFI) program, a plan for developing and implementing this program, and a plan for centralized management, by NRR, of the FPFI program and all other reactor fire protection work. The principal objective of these efforts was to ensure that the NRC has a strong, broad-based, and coherent fire protection program that is commensurate with the issue’s safety significance.

Conclusion

The staff developed the FPFI program inspection procedures and guidance and drafted recommendations for centrally managing all reactor fire protection reviews and inspections, using Headquarters and regional staff qualified to perform such work. With contract assistance from Brookhaven National Laboratory, the staff continued to develop a probabilistic risk assessment model for the self-induced station blackout study. Brookhaven National Laboratory also drafted a report on risk-based approaches for evaluating fire mitigation features in nuclear power plants. Thus, this issue addressed the staff’s efforts to improve its capability to make independent assessments of safety and, therefore, was considered to be a licensing issue.[1731]

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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1601. Memorandum for C. Serpan from A. Chaffee, “Nuclear Reactor Regulation (NRR) Input into Research NUREG-0933 (WITS Item 9400213),” February 13, 1996. [9602260124]

1626. Memorandum for A. Thadani from G. Holahan, “Revision to Report on the Re-Assessment of the NRC Fire Protection Program,” February 27, 1993. [9504190319]

1627. SECY-93-143, “NRC Staff Actions To Address the Recommendations in the Report on the Reassessment of the NRC Fire Protection Program,” U.S. Nuclear Regulatory Commission, May 21, 1993. [9306030231]

1628. SECY-95-034, “Status of Recommendations Resulting from the Reassessment of the NRC Fire Protection Program,” U.S. Nuclear Regulatory Commission, February 13, 1995. [9503060019]

1629. Memorandum for Chairman Jackson et al. from J. Taylor, “Semiannual Report on the Status of the Thermo-Lag Action Plan and Fire Protection Task Action Plan,” September 20, 1995. [9509250375]

1731. Memorandum for W. Russell from D. Morrison, “Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996,” June 24, 1996. [9606260260]

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ISSUE 199: IMPLICATIONS OF UPDATED PROBABILISTIC SEISMIC HAZARD ESTIMATES IN CENTRAL AND EASTERN UNITED STATES ON EXISTING PLANTS

Description

Historical Background

On May 26, 2005, the Office of Nuclear Reactor Regulation (NRR), Division of Engineering, recommended that issues related to a closed generic seismic issue (Generic Issue (GI)-194, "Implications of Updated Probabilistic Seismic Hazard Estimates," dated September 23, 2003) and the impact of higher seismic hazard on current nuclear power plants (NPPs) in the Central and Eastern United States (CEUS) region be examined under the GI identification and resolution process.[1930] On June 9, 2005, GI-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States," joined the list of GIs.[1931]

Safety Significance

Recent data and models indicate that estimates of the potential for earthquake hazards for some NPPs in the CEUS may be larger than previous estimates. While it has been determined that currently operating plants remain safe, the recent seismic data and models warrant further study and analysis. This further analysis will allow the U.S. Nuclear Regulatory Commission (NRC) to better understand the current margins at plants for earthquakes.

Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion,"[1932] developed in the early 1990s, specifies a reference probability for exceedance of a safe-shutdown earthquake (SSE) ground motion (i.e., seismic hazard) at a median annual value of 1E-5. This reference probability value is based on the annual probability of exceeding the SSEs for 29 CEUS nuclear power sites and is used to establish the SSEs for future nuclear facilities. Based on preliminary results from work performed by the United States Geological Survey (USGS) in 2004, it appears that the reference probability for the 29 CEUS sites has increased to about 6 to 7E-5. The increase in the reference probability value is primarily due to recent developments in the modeling of earthquake ground motion in the CEUS. When the staff first identified this issue, no new plants had applied for a construction permit or early site permit (ESP) since Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," was revised and Regulatory Guide 1.165[1932] was issued in 1997. When the staff began review of the ESP applications, the staff realized the impact of the revised regulation and the regulatory guide as they relate to future plants and operating reactors.

From the staff's review of the ESP applications with support from the 2004 USGS draft report, it appeared that the perception of seismic hazard for operating plants in the CEUS region had increased. Based on the evaluations of the Individual Plant Examination of External Events (IPEEE) Program, the staff had determined that seismic designs of operating plants in the CEUS provided an adequate level of protection. However, in light of the preliminary results from the USGS work of 2004 and the ESP applications, the staff also recognized that the probability of exceeding the SSE at some of the currently operating sites in the CEUS is higher than previously understood. Therefore, the staff initiated this GI to assess the impact of increased estimates of seismic hazards on selected current NPPs in the CEUS region that might be impacted by the updated seismic research, information, and models.

Screening Analysis

In December 2007, the staff completed the screening analysis using guidance contained in Management Directive 6.4, “Generic Issues Program,”[1858] and SECY-07-0022, “Status Report on Proposed Improvements to the Generic Issues Program,” dated January 30, 2007.[1888] The screening panel reviewed the analysis in January 2008. On February 1, 2008, the Director of the Office of Nuclear Regulatory Research (RES) approved the screening panel recommendation[1933] to begin the safety/risk assessment stage of the generic issue process.

The screening panel’s recommendation was based on the screening analysis, which showed that GI-199 passed the seven GI screening criteria. The discussion under each criterion below provides the screening analysis for GI-199.

1. The issue affects public health and safety, the common defense and security, or the environment.

The estimated risk to public health and safety and the environment associated with the occurrence of seismic events at some NPP sites might have increased from previous estimates. The issue stems from ongoing research being conducted by a number of scientists into the seismic history of the CEUS and the details of wave propagation and attenuation in this region. In particular, information submitted to the NRC by ESP applicants contained updated seismic information that included new models to estimate earthquake ground motion and updated models for earthquake sources in seismic regions such as eastern Tennessee and around both Charleston, South Carolina, and New Madrid, Missouri. In addition, information summarized by the USGS as part of the National Seismic Hazard Mapping Program indicates that the estimated likelihood of seismic activity (i.e., seismic hazard) in some CEUS locations has increased from previous estimates. Some of these locations are near existing NPP sites. An increase in the seismic hazard at these sites has the potential to adversely impact public health and safety if the estimated increased seismic hazard were to significantly exceed plant design capabilities; substantially reduce perceived safety margins for plant structures, systems, and components (SSCs) important to safety; or appreciably increase the risk associated with the plant’s response to a seismic event. From a qualitative perspective, if the increased hazard is significant at sites that have relatively small safety margins for seismic events, then the estimated risk for these sites could increase.

2. The issue applies to two or more facilities and/or licensees/certificate holders or holders of other regulatory approvals.

The updated information described above results in increased estimates of the seismic hazard that could occur at multiple, although not all, NPP sites in the CEUS. Specifically, updated models for earthquake sources in seismic regions such as eastern Tennessee and around both Charleston, South Carolina, and New Madrid, Missouri, indicate that the rate of earthquake occurrence in these regions is greater than previously recognized. Because this change applies to several large regions, it has the potential to affect more than one NPP site. Further, new models used to estimate earthquake ground motion have been revised relative to those used in the 1980s. This change also has the potential to affect more than one NPP site. Updated estimates of seismic hazard values at some of the sites could potentially exceed the design basis as well as the review-level earthquake spectrum used as part of the IPEEE Program.

3. The issue cannot be readily addressed through other regulatory programs and processes; existing regulations, policies, or guidance; or voluntary industry initiatives.

In a memorandum to RES dated May 26, 2005, NRR identified this issue and recommended that it be examined under the Generic Issues Program.[1934] In this memorandum, the staff concluded that the seismic designs of operating plants in the CEUS still provide adequate safety margins while the staff continues to evaluate new seismic hazard data and models and their potential impact on plant risk estimates. At the same time, the staff also recognized that these new seismic data and models could reduce available safety margins due to increased estimates of the probability associated with seismic hazards at some of the currently operating sites in the CEUS. Therefore, to help assess the potential reduction in available safety margins using a probabilistic approach, the NRR staff recommended that the new data and models on CEUS seismic hazards be examined under the Generic Issues Program.[1934] Accordingly, at that time, the NRR staff determined that this issue was not sufficiently characterized to be addressed under existing licensing processes for licensees of plants that might be impacted.

Based on the limited evaluation of available information, this issue does not appear to be adequately characterized for complete treatment under existing regulatory programs and processes. Examples of regulatory programs and processes that might apply after obtaining additional information and performing further evaluations are listed below. Additional analysis will help determine whether this issue is amenable to these or other regulatory programs or industry initiatives.

- LIC-100, “Control of Licensing Bases for Operating Reactors”
- LIC-105, “Managing Regulatory Commitments Made by Licensees to the NRC”
- LIC-202, “Procedures for Managing Plant-Specific Backfits and 50.54(f) Information Requests”
- LIC-300, “Rulemaking Procedures”
- LIC-400, “Procedures for Controlling the Development of New and Revised Generic Requirements for Power Reactor Licensees”
- LIC-401, “NRR Reactor Operating Experience Program”
- LIC-501, “Program Coordination for Risk-Informed Activities”
- LIC-503, “Generic Communications Affecting Nuclear Reactor Licensees”
- LIC-504, “Integrated Risk-Informed Decision-Making Process for Emergent Issues”

4. The issue can be resolved by new or revised regulation, policy, or guidance.

Further analysis of the risk or safety impact would provide sufficient additional information to properly characterize the issue and its potential impact on CEUS plants and support consideration under other existing regulatory programs or industry initiatives. The regulatory office has authority to take appropriate regulatory action(s) as necessary to protect the public health and safety and the environment. Depending on the outcome of the additional analysis, as well as industry initiatives to address any safety issues, the regulatory office could address this issue through one or more actions involving regulation, policy, or guidance.

5. The issue’s risk or safety significance can be adequately determined (i.e., it does not involve phenomena or other uncertainties that would require long-term studies and/or experimental research to establish the risk or safety significance).

The screening analysis was performed based on the staff's review of updated seismic data and models submitted by ESP applicants and also updated seismic hazard data and models available from the USGS as part of the National Seismic Hazard Mapping Program. The seismic hazard at CEUS plant sites of interest can be evaluated using an approach like the detailed assessment performed by the Electric Power Research Institute (EPRI)[1935] for 28 of the 29 sites included in Regulatory Guide 1.165.[1932] This study used updated attenuation models and incorporated updates to the EPRI seismic source model developed during the preparation of the ESPs. The risk significance of the updated seismic hazard information can be evaluated for CEUS plant sites of interest by performing a comparison of uniform hazard spectra or other hazard results to the beyond-design-basis review-level earthquake or hazard curve used as part of the IPEEE evaluation.[1798] The available IPEEE Program results would allow a general assessment of the potential safety impact of increases in seismic hazard at specific sites. This analysis was performed later as part of the safety/risk assessment under the Generic Issues Program.

6. The issue is well defined, discrete, and technical.

The seismic hazard will be adequately defined upon detailed assessment of available updated seismic data and models submitted by ESP applicants and also updated seismic hazard data and models available for other CEUS plant sites of interest using an approach like that performed by EPRI[1935] for 28 of the 29 sites included in Regulatory Guide 1.165.[1932] This will allow the seismic hazard estimates for CEUS plant sites of interest to reflect the state of current knowledge. As new information and research becomes available, future updates might be warranted. The plants' response to seismic hazards involves technical analyses using established techniques.

7. Resolution of the issue may potentially involve review, analysis, or action by the affected licensees, certificate holders, or holders of other regulatory approvals.

After further characterization of site-specific seismic hazards and an analysis of the plant's response to the increased seismic hazard, some plants may be identified as having a vulnerability that must be addressed to maintain adequate safety margins. Determining a plant's margin and potential need for action to maintain an adequate margin could involve regulatory actions (e.g., requests for information from plant licensees, reviews, additional analysis, mitigation actions, physical enhancements, administrative controls) for some plant licensees or could involve actions by industry stakeholders.

The screening analysis showed that the estimated increase in spectral acceleration for some existing CEUS plant sites might exceed the design basis and values used for the NRC's review of IPEEE submittals. This translates into an equivalent increase in seismic demand on plant SSCs. As a result, this issue has the potential to result in increased seismic core damage frequency (SCDF) estimates for some plants. However, the screening analysis provided a limited evaluation that did not assess the safety response of the plants.

The limited scope screening analysis concluded that the seismic designs of operating plants in the CEUS provided adequate safety margins while the staff continued to evaluate new seismic hazard data and models and their potential impact on plant risk estimates. Specific reasons for this conclusion included the following:

- The estimated annual probability of exceedance of seismic hazard is small in an absolute sense.

- Earthquakes cause ground motion over a range of frequencies. Lower frequency motions are more damaging to buildings and equipment than higher frequency motions. Based on the NRC staff's reviews associated with ESPs, the staff was confident that the recent seismic data and models would show that increased estimates of the seismic hazards would occur primarily in the higher ground motion frequencies. Accordingly, the staff anticipated that these increased estimates of seismic hazards would primarily have little impact on previous estimates of the potential damage to buildings and equipment.

- The plants are designed to withstand anticipated earthquakes with substantial design margins. Plants may have seismic margins beyond those reflected in their IPEEE submittals, and these could compensate for the increase in estimated seismic load. Such additional seismic margins at plants may be inherent in the design and construction, realized from improved data and analysis methods, or result from plant modifications or enhancements completed since the IPEEE submittals.

Based on the knowledge of this issue at the time of the screening analysis and its potential effect on CEUS plants, this issued passed the seven GI screening criteria and, therefore, warranted further analysis under the Generic Issues Program.

Safety/Risk Assessment

RES staff developed and implemented a methodology to determine the implications of updated probabilistic seismic hazard estimates in the CEUS on existing plants. The methodology, analyses, results, and limitations of the safety risk assessment are summarized below. A detailed discussion of the safety/risk assessment is documented in the NRC's "Safety/Risk Assessment Results for Generic Issue 199, Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plant: Safety/Risk Assessments," dated August 2010. [1974]

Risk Methodology

SCDF was chosen as the appropriate risk metric because it is expected to be more sensitive than other metrics (either large early release fraction or public dose) to changes in the seismic hazard. In addition, SCDF can be estimated using IPEEE information. Conversely, the IPEEE Program did not produce sufficient quantitative information to estimate alternate risk metrics.

The staff performed a two-stage assessment to determine the implications of updated probabilistic seismic hazards in the CEUS on existing NPPs. The first stage involved evaluating the change in seismic hazard with respect to previous estimates at individual plants. The second stage estimated the change in SCDF as a result of the change in the seismic hazard for each operating plant in the CEUS. The seismic hazard at each NPP site depends on the unique seismology and geology surrounding the site, which necessitated separately determining the implications of updated probabilistic seismic hazard for each of the 96 operating NPPs in the CEUS.

Evaluation of Changes in Seismic Hazard Estimates

In the first stage of the assessment, the NRC staff evaluated the potential significance of changes in seismic hazards in a stepwise fashion by assessing the degree to which the seismic hazard estimates developed using the most recent seismic hazard information and NRC staff guidance deviate from previously developed assessments. The comparison of results indicated an increase in the seismic hazard estimates relative to previous assessments for a number of plants.

Evaluation of Changes in Seismic Core Damage Frequency

In the second stage, the NRC staff developed SCDF estimates using three sets of mean seismic hazard curves (the 1989 EPRI study, the 1994 Lawrence Livermore National Laboratory study, and a 2008 USGS study) and plant-level fragility curves developed from information provided in the IPEEE submittals. This method had previously been used by the staff in the resolution of GI-194 and during reviews of various risk-informed license amendments.[1975] The changes in the NRC's SCDF for a number of plants lie in the range of 10^{-4} per year to 10^{-5} per year, which meets the numerical risk criteria for an issue to proceed to the regulatory assessment phase of the Generic Issues Program.

Overall seismic risk estimates remain small in an absolute sense. All operating plants in the CEUS have a SCDF less than or equal to 10^{-4}/year, confirming that there is no immediate concern for adequate protection.

The approach used to estimate SCDF in the safety/risk assessment does not provide insight into which SSCs are important to seismic risk. Such knowledge provides the basis for postulating plant backfits and conducting a value/impact analysis of potential backfits during a regulatory analysis. For a number of plants, especially those that performed reduced-scope seismic margin analysis, detailed information is presently not available to the NRC regarding plant seismic capacity (the ability of a plant's SSCs to successfully withstand an earthquake) beyond the required design-basis level.

Safety/Risk Assessment Panel Conclusions and Observations

In accordance with Management Directive 6.4, a safety/risk assessment panel was established to determine, on a generic basis, if the risk associated with GI-199 warranted further investigation for potential imposition as a cost-justified backfit and to provide a recommendation for the next step.

The panel completed its independent review of the safety/risk assessment for GI-199 in September 2010.[1976] The panel reached the following conclusions and observations:

- Overall seismic core damage risk estimates are consistent with the Commission's safety goal policy statement because they are within the subsidiary objective of 10^{-4}/year for core damage frequency. The GI-199 safety/risk assessment, based in part on information from the NRC's IPEEE Program, indicates that no concern exists about adequate protection and that the seismic design of operating reactors provides a safety margin to withstand potential earthquakes exceeding the original design basis.

- The changes in SCDF estimated in the safety/risk assessment stage of GI-199 for numerous plants lie in the range of 10^{-4}/year to 10^{-5}/year, which meet the numerical risk criteria for an issue to proceed to the regulatory assessment stage of the generic issues program.

- New consensus seismic-hazard estimates will become available in 2011 (these are a product of a joint NRC, U.S. Department of Energy, USGS, and EPRI project). These consensus seismic hazard estimates will supersede the existing EPRI, Lawrence Livermore National Laboratory, and USGS hazard estimates used in the GI-199 safety/risk assessment.

- Certain factors that affect the development of realistic SCDF estimates will remain unresolved even after the new consensus seismic hazard estimates are developed. The issue is primarily that many IPEEEs did not produce SCDF estimates and so lack some of the information needed to produce such estimates.
    - For a number of the plants that performed reduced-scope seismic margin analyses as part of the IPEEE Program, limited detailed information exists about plant seismic capacity (the ability of a plant's SSCs to successfully withstand an earthquake) beyond the required design-basis level.
    - The approach used in the safety/risk assessment to estimate SCDF considered the plant-level seismic capacity and, therefore, did not provide insight into which SSCs were important to seismic risk. Such knowledge would be required in order to postulate potential cost-beneficial backfits.
- IPEEE submittals generally provided limited, qualitative information about the seismic capability of containments. Any regulatory analysis of GI-199 should consider potential plant modifications for reducing the probability of seismically induced containment failure as discussed in Section 3.3.1 of NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 4, issued September 2004. [1977]

Conclusion

The panel recommended transferring the lead responsibility for subsequent GI-199 actions to NRR for regulatory office implementation and taking further actions to address GI-199 outside the Generic Issues Program (i.e., obtain information and develop methods, as needed, to complete plant-specific value/impact analyses of potential backfits to reduce seismic risk).

The NRC issued information notices in September 2010 to inform stakeholders of the issuance of the GI-199 safety and risk assessment report. Information Notice 2010-18, "Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" dated September 2, 2010, [1978] was issued to NPPs and independent spent fuel storage installations (ISFSI). It stated that the NRC will follow the appropriate regulatory process to request operating plants and ISFSIs to provide specific information relating to their facilities to enable the NRC staff to complete the regulatory assessment where candidate backfits are identified and evaluated. Information Notice 2010-19, "Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States," dated September 16, 2010, [1979] was issued to fuel cycle facilities. GI-199 is in the regulatory office implementation stage, in accordance with Management Directive 6.4.[1858]

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1978. Information Notice 2010-18, “Generic Issue 199, ‘Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,’” U.S. Nuclear Regulatory Commission, September 2, 2010. [ML101970221]

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TASK HF2: TRAINING

In response to Section 306 of the Nuclear Waste Policy Act of 1982, the Commission published a "Policy Statement[966] on Training and Qualification of Nuclear Power Plant Personnel, March 1985. This Policy Statement endorsed the INPO Accreditation Program as a means of ensuring that the five essential elements of performance-based training, as described in the Policy Statement, are met in industry training. The continuation of this endorsement is based on the suc-cess of industry programs after a 2-year period during which the NRC will con-tinue to evaluate applicant and licensee implementation of improvement programs. Revisions to SRP[11] Section 13.2 will also be prepared.

Two new inspection procedures have been completed and issued for regional use: IP 41701, "Licensed Operator Training," and IP 41400, "Nonlicensed Staff Training." These procedures emphasize evaluation of licensee staff performance and exclusively cover training areas subject to INPO accreditation. This task was identified as three distinct items in Table 7 of the NRC 1985 Annual Report (Items 2.1, 2.2, and 2.3). The following is a discussion of these three items.

ITEM HF2.1: EVALUATE INDUSTRY TRAINING

Description

The staff published NUREG-1220[993] to be used for post-accreditation review of those training programs included in the Policy Statement.[966] While the purpose of these reviews was to evaluate the thoroughness of the INPO accreditation procedures, the criteria also would enable the staff to conduct independent evaluations of nuclear utility training programs. In addition to these reviews, the evaluation of industry training included such criteria as inspection reports, license examination results, event-based reviews, and SALP reports for both accredited and non-accredited programs.

In the Supplement to NUREG-0933 published in 1986, staff stated that the NRC effort on this item was ongoing and would be reassessed after the 2-year evaluation period. This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM HF2.2: EVALUATE INPO ACCREDITATION

Description

In addition to the evaluation of industry training programs using NUREG-1220,[993] NRC observers accompany INPO evaluation teams during their review of utility training programs to determine whether the INPO teams conduct a thorough review of training programs using the INPO criteria and objectives for accreditation. The staff also observes the meetings of the National Nuclear Accrediting Board to determine its effectiveness in making decisions to accredit industry programs.

In the Supplement to NUREG-0933 published in 1986, staff stated that the NRC effort on this item was ongoing and would be reassessed after a 2-year period. This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM HF2.3: REVISE SRP SECTION 13.2

Description

In the Supplement to NUREG-0933 published in 1986, staff stated that the staff would revise SRP[11] Section 13.2 to be consistent with the Commission policy to encourage performance-based training in nuclear power plants. This effort was ongoing and would be reassessed when Items HF2.1 and HF2.2 and final rulemaking for revisions to operator licensing (10 CFR 55 and conforming amendments) were completed.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated

November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

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TASK HF4: PROCEDURES

This task was developed to provide assurance that plant procedures were adequate and could be used effectively. The objective was to provide procedures that would guide operators in maintaining plants in a safe state under all operating conditions, including the ability to control upset conditions without first having to diagnose the specific initiating event. This objective was to be met by: (1) developing guidelines for preparing, and criteria for evaluating, EOPs, normal operating procedures, and other procedures that affect plant safety; and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures. This task was divided into five distinct items as discussed below.

ITEM HF4.1: INSPECTION PROCEDURE FOR UPGRADED EMERGENCY OPERATING PROCEDURES

Description

Criteria to evaluate and inspect EOPs by the regions were prepared by NRR and OIE and published as an OIE Temporary Instruction (TI). Similar criteria and inspection modules were to be developed when the guidelines for the upgrading of other procedures were completed.

In December 1982, Supplement 1 to NUREG-0737[98] was issued as Generic Letter 82-33[376] with a requirement for each plant to submit a Procedures Generation Package (PGP) as a part of the effort to upgrade EOPs; the generic letter also indicated that the NRC would audit upgraded EOPs on a selective basis. In 1984, the NRC began auditing upgraded EOPs. After conducting several audits, the staff issued Information Notice No. 86-64[1210] to advise the industry that there were indications that many licensees were not appropriately developing and implementing upgraded EOPs. Based on the deficiencies identified in the Information Notice, the staff concluded that some licensees might not have appropriately developed and implemented upgraded EOPs in accordance with their PGPs. The staff decided to: (1) continue with its audit program to further determine the scope and safety significance of the deficiencies identified in the Notice;[1210] and (2) conduct inspections at all plants to evaluate the implementation of licensee commitments to develop and implement upgraded EOPs.

Conclusion

This issue was given a high priority ranking and pursued by the staff. In June 1986, the staff prepared TI 2515/79 which contained criteria for inspecting how well licensees were complying with their PGP commitments. In April 1987, the staff issued a supplement to its first Information Notice[1210] based on evaluations from 6 additional plants. In early 1988, the staff suspended its program to evaluate licensees' compliance with programmatic requirements (i.e., PGPs) and redirected its efforts to focus more on the technical adequacy and useability of the EOPs. Lessons learned by the staff from its inspection program for EOPs were published in NUREG-1358.[1209] TI 2515/92,[1209] "Emergency Operating Procedures Team Inspections," contains guidance for conducting these inspections. Based on the results from this inspection program of 28 plants, NRR was responsible for developing a program of inspections for the remaining plants. Thus, this issue was RESOLVED and no new requirements were established.[1208]

ITEM HF4.2: PROCEDURES GENERATION PACKAGE EFFECTIVENESS EVALUATION

Description

To evaluate the effectiveness of the NRC's long-term program for upgrading EOPs, the staff audited the implementation of PGPs at selected plants. PGPs describe a plant's program for adapting generic technical guidelines to develop the technical content of plant-specific EOPs and applying human factors principles to produce EOPs that are useable by operators. Six audits were performed and additional audits were planned before an assessment of the program was completed.

Based on input from sources including PGP implementation audits, staff PGP reviews, and license examiners, the staff identified problems that plants were experiencing with implementing their PGPs. To alert the industry to these problems, the staff issued Information Notice No. 86-64.[1210] Progress by the industry in addressing the problems identified in the Notice were to be monitored by inspections, additional PGP implementation audits, and through continued dialogue with the industry.

Conclusion

This item was related to increasing the staff's knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety and, therefore, was classified as a Licensing Issue.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM HF4.3: CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS

Description

A safety evaluation standard was to be developed to screen licensee proposals to place additional burdens upon operators. Licensees proposing to resolve severe accident issues or other generic safety issues by adding to EOPs and training, in lieu of hardware fixes, were expected to utilize the standard to verify that the additional burdens placed upon operators did not overload the operators, and that the additional operator responsibilities were adequately covered in procedures and training. This standard was to be applied to any licensee proposing to add additional operator responsibilities as part of the resolution of a generic safety issue; the staff did not anticipate that it would be applied retroactively to DBAs or existing EOPs. The standard would not impose requirements upon plant design or operation directly, but could narrow the range of options available to resolve other issues. The likely form of the standard was believed to be an SRP[11] Section.

Conclusion

This item was covered in Issue B-17.

ITEM HF4.4: GUIDELINES FOR UPGRADING OTHER PROCEDURES

Description

On the basis of efforts to evaluate the quality of, and the problems associated with, existing plant procedures, NRR evaluated the need to develop technical guidance for use by the industry in upgrading normal operating procedures and abnormal operating procedures (AOPs) similar to what the staff completed for EOPs. The staff was to perform a regulatory analysis to determine whether regulatory action for other plant procedures was warranted and, if so, develop formal regulatory requirements.

Conclusion

This issue was given a high priority ranking and RESOLVED with no new requirements.[1562] The staff prepared a summary of good practices that licensees could use in performing any voluntary upgrade of procedures.[1576] In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

ITEM HF4.5: APPLICATION OF AUTOMATION AND ARTIFICIAL INTELLIGENCE

Description

The level of automation possible within the nuclear industry spans a range of possibilities from the fully manual, with locally operated valves, to the fully automated, employing artificial intelligence (AI). The nuclear industry is basically at the one-switch one-valve end of that range. The reliability of AI for safety-related uses was unproven; however, evidence from other industries suggested that there can be significant savings in operating costs as well as an enhancement in safety with increased automation of operator actions. Reducing the menial level workload of operators could provide better low-level control and fewer operator errors. Such automation can also free operators to concentrate on the cognitive level of operations. The subject of automation and AI affects control room design, operating procedures, and other operator aids, staffing, and training. The staff was to investigate the benefits and hazards of increased automation in the nuclear industry and consider incentives to encourage the movement toward automation as a means of increasing plant safety.

Conclusion

This item was covered in Item HF5.2.

References

11. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, (1st Ed.) November 1975, (2nd Ed.) March 1980, (3rd Ed.) July 1981.

98. NUREG-0737, “Clarification of TMI Action Plan Requirements,” U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

376. Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits from U.S. Nuclear Regulatory Commission, “Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (Generic Letter No. 82-33),” December 17, 1982. [8212060349]

1208. Memorandum for V. Stello from T. Murley, “Final Resolution of Generic Issue (GI) HF4.1, Inspection Procedure for Upgraded Emergency Operating Procedures,” October 17, 1988. [8811070169]

1209. NUREG-1358, “Lessons Learned from the Special Inspection Program for Emergency Operating Procedures,” U.S. Nuclear Regulatory Commission, April 1989.

1210. Information Notice 86-64, “Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures,” U.S. Nuclear Regulatory Commission, August 14, 1986 [8608120028], (Supplement 1) April 20, 1987 [8704160062].

1562. Memorandum for J. Taylor from E. Beckjord, “Resolution of Human Factors Generic Issue 4.4, ‘Guidelines for Upgrading Other Procedures,’” July 29, 1993. [9502070331]

1564. Memorandum for W. Russell from E. Beckjord, “License Renewal Implications of Generic Safety Issues (GSIs) Prioritized and/or Resolved Between October 1990 and March 1994,” May 5, 1994. [9406170365]

1576. “Approaches to Upgrading Procedures in Nuclear Power Plants,” Pacific Northwest Laboratory, August 1994. [9507280167]

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

TASK HF7: HUMAN RELIABILITY

The primary purposes of this task are to develop a technical support system for NRC reliability evaluations, especially the PRA programs, and to provide feedback links from operating experience to other elements of the human factors program. A secondary goal is to develop approaches for employing human error data as baseline performance measures in man-machine safety system evaluations.

ITEM HF7.1: HUMAN ERROR DATA ACQUISITION

Description

Staff stated in the Supplement to NUREG-0933 published in 1986 that ongoing and planned activities were designed to provide NRC reliability evaluation programs with methods and techniques for acquiring reliable human error data from a variety of nuclear power related sources. Significant research involved developing guidelines for acquiring human error data from expert judgment, training simulators, operating nuclear power plants using LER data, and from a non-punitive reporting concept.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM HF7.2: HUMAN ERROR DATA STORAGE AND RETRIEVAL

Description

Staff stated in the Supplement to NUREG-0933 published in 1986 that activities were designed to provide the NRC with a human reliability data bank for use in processing human error data for use by reliability evaluation specialists. Planned activities included developing methods and procedures for computing human error probability statements from diverse information sources and storing, updating and retrieving human error probability statements and related information.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM HF7.3: RELIABILITY EVALUATION SPECIALIST AIDS

Description

Staff stated in the Supplement to NUREG-0933 published in 1986 that that a comprehensive and accurate analysis of human behavior sequences leading to recognition, diagnosis and reaction to nuclear power station normal, transient and emergency events was necessary for risk assessment. Analytic techniques and methods for portraying adequately the human segments of those events were needed, especially events involving redundant or interdependent actions by individuals or groups. Also needed were techniques for analyzing cognitive and performance shaping factor (e.g., stress, fatigue, attitude) aspects of human behavior. Significant research activities in this area involved: (1) developing techniques for analyzing safety-related events, especially those involving redundancy and/or interdependent actions; and (2) investigating the feasibility of objectively analyzing cognitive and performance shaping aspects of human behavior within the content of NRC reliability evaluation programs, especially PRAs.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM HF7.4: SAFETY EVENT ANALYSIS RESULTS APPLICATION

Description

The PRAs are a potential source of quantitative and qualitative human performance data, both generic and plant-specific. Staff stated in the Supplement to NUREG-0933 published in 1986 that human reliability research would be directed toward developing and testing approaches and techniques for systematically using human performance data from PRAs to: (1) identify generic and plant-specific man-man and man-machine safety system retrofit requirements, (2) establish objective baseline performance measures for evaluating plant retrofits, and (3) identify future human reliability/human factors research needs.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

TASK CH1: ADMINISTRATIVE CONTROLS AND OPERATIONAL

This task, outlined in Chapter 1 of NUREG-1251,[1174] called for the staff to review the administrative controls over plant operations in the U.S. to determine if adequate controls are in place to maintain plant conditions within the safe operating envelope. This review will include an assessment of procedural adequacy and compliance, approval of tests, bypassing of safety systems, availability of engineered safety features (ESF), operating staff attitudes toward safety, management systems, and accident management.

ITEM CH1.1: ADMINISTRATIVE CONTROLS TO ENSURE THAT PROCEDURES ARE FOLLOWED AND THAT PROCEDURES ARE ADEQUATE

This item consists of two recommendations that are evaluated separately below.

ITEM CH1.1A: SYMPTOM-BASED EOPs

Description

During the Chernobyl event, serious operational errors aggravated the emergency situation that existed and were considered to be a major contributor to the disastrous consequences that ensued. Although design and operational control protections at U.S. reactors provide assurance against the chain of events that occurred at Chernobyl, the Chernobyl experience suggests that closer attention should be paid to effective emergency procedures and the ability of operators to use them. Symptom-based EOPs and their full implementation are a key part of the necessary preparedness for effective management of emergencies. Recent audits by the NRC have identified deficiencies in the implementation of the new symptom-based EOPs. In addition, NRC examinations have identified the need for additional training in the use of these EOPs. The staff has undertaken an accelerated inspection program of EOPs which is aimed at evaluating their technical correctness and their ability to be physically and correctly carried out. This program consists of a four-team effort encompassing four units of each of the four reactor vendor types. Possible regulatory action to upgrade this program or possible further study of any inconclusive results will be considered following staff review of the results of this inspection program.

Staff stated in the Supplement to NUREG-0933 published in 1989 that this issue was directed toward integration of Chernobyl lessons into the staff's EOP effort and was expected to increase the staff's knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, it was considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated

November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.1B: PROCEDURE VIOLATIONS

Description

Procedure violations at nuclear power plants are committed by licensed and auxiliary operators, plant technicians, maintenance personnel, and contractors. While the NRC believes that these violations are infrequent and only rarely occur with the knowledge that they are being committed, the exact nature and extent of these violations and their consequences are basically unknown. At Chernobyl, serious procedure violations were a key factor in the cause of the accident. This issue called for the staff to identify procedure violations committed at nuclear power plants, evaluate their consequences, and, if warranted, recommend options for regulatory actions to minimize future violations. The staff will focus initially on those procedure violations associated with reactor scrams or scram signals and will address the following:

(a) Analyze incident reports and other descriptions of major events and identify procedure violations that contributed to initiation of the events or that occurred during the events.

(b) Conduct a literature search for other sources of documented procedure violations associated with reactor scrams or scram signals.

(c) Review the special study AEOD/S8011176 for incidences of procedure violations.

(d) Develop Sequence Coding and Search System (SCSS) search criteria and review LERs for reports of procedure violations. The LER search will be limited to the period 1983 to the present.

(e) Analyze the above data and develop and implement an approach for their presentation that will provide: (1) the kinds of procedure violations and the personnel involved; (2) the frequency of procedure violations involving reactor scrams; (3) the consequences of these violations, including challenges to ESF, and actual or potential releases of radioactive materials; and (4) the frequency of procedure violations with significant consequences.

Staff stated in the Supplement to NUREG-0933 published in 1989 that in pursuing this issue, the staff was expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue was considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated

November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.2: APPROVAL OF TESTS AND OTHER UNUSUAL OPERATIONS

This item consists of two recommendations that are evaluated separately below.

ITEM CH1.2A: TEST, CHANGE, AND EXPERIMENT REVIEW GUIDELINES

Description

Planned tests and experiments not described in licensees' SARs and changes to facilities and procedures described in these reports are required to be evaluated beforehand by licensees, in accordance with 10 CFR 50.59, to assure their safety and that the NRC is afforded the opportunity to review them where appropriate. Thousands of these reviews are successfully conducted by licensees each year; however, in some instances, these reviews have not been adequate. As a result, the NRC was not always afforded the opportunity to review those tests, experiments, and changes that involved an unreviewed safety question before they were performed. Without appropriate reviews by licensees and the NRC, tests could be performed without adequate safety provisions or some safety features could be unacceptably altered, a condition that could remain undetected for lengthy periods. The Chernobyl accident occurred during a test and the lack of adequate planning review, preparation, and implementation of the test emphasizes the need for attention to this issue.

The need for review guidance for tests, changes, and experiments was identified before the Chernobyl accident and is being addressed by a NUMARC/NSAC Working Group and by the NRC Technical Specifications Branch in the Technical Specifications Improvement Program (TSIP). Staff stated in the Supplement to NUREG-0933 published in 1989 that the NUMARC/NSAC Working Group would develop draft criteria and guidelines and provide them to the industry and the NRC for review and comment. When acceptable to the Working Group and a consensus of the industry agreed, the NRC would review the guidance document which would be made available to all licensees and might be supplemented if necessary to permit NRC endorsement. The industry and the NRC would use the guidance in their review of tests, experiments, and changes required by 10 CFR 50.59. The scope of this issue was limited to coordination to assure appropriate introduction of Chernobyl lessons into the ongoing program.

In pursuing this issue, the staff was expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue was considered to be a licensing issue.

Conclusion

An NRC Working Group consisting of seven members and two ad hoc members was formed in July 1987 to coordinate with NUMARC/NSAC the development of guidance for 10 CFR 50.59 reviews including tests, experiments, and changes and to recommend an endorsable product to NRC management. Regional coordinators were named to interact with the Working Group and to assist it in various requests, including comment requests on NUMARC/NSAC draft documents. Two drafts of the NUMARC/NSAC Working Group "10 CFR 50.59 Guidance Document" were forwarded to the NRC for comment.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.2B: NRC TESTING REQUIREMENTS

Description

There is a potential for human error when conducting tests to assess equipment capabilities. This potential represents a risk to plant safety which can vary in severity depending both on the nature of the tests and the circumstances associated with them. Tradeoffs between the risks of not testing or of testing at a lesser frequency and the risks associated with such testing have not always been assessed. The Chernobyl accident occurred when the unit was used for a test. This issue called for the staff to determine if there are any post-startup equipment tests whose conduct presents a sufficient potential impact on plant safety to suggest either their modification, reduced frequency, or elimination.

Staff stated in the Supplement to NUREG-0933 published in 1989 that the staff would review NRC-required post-initial-startup equipment tests at nuclear power plants to identify those tests where human error could result in risks to plant safety. For this issue, "risk to plant safety" was defined as a reactor scram or scram signal, a challenge to ESF, unanticipated releases of radioactive materials, or any other evident unacceptable plant condition. The staff would quantify the potential risk for such tests and recommend a revised testing requirement for those with excessive risk. In resolving this issue, the staff will:

(a) Devise search criteria and conduct a search of the SCSS data bank of LERs to identify reported cases of human error associated with the conduct of plant equipment tests. The search will cover the period 1984 to the present.

(b) Screen the LER data collected to identify for further study those errors that resulted in reactor trips, challenges to ESF, unanticipated releases of radioactivity, or other evident unacceptable plant conditions. The objective is to order the LERs in terms of their results and to screen out those human errors, e.g., failure to conduct a test on time, which have no immediate consequence potential.

(c) Conduct a literature search for other analyses or descriptions of human error and resulting non-trivial consequences associated with plant testing.

(d) Using the above data, prepare a preliminary estimate of the potential risk to plant safety caused by human error during equipment testing. This estimate should support a recommendation to terminate this issue or to continue with more detailed risk/benefit analyses that could provide additional scope to the Performance Evaluation of Technical Specifications (PETS) program or support revisions to NRC testing policy.

In pursuing this issue, the staff was expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue was considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.3: BYPASSING SAFETY SYSTEMS

This item consists of one recommendation that is evaluated below.

ITEM 1.3A: REVISE REGULATORY GUIDE 1.47

Description

The bypassing of safety functions by inadvertently bypassing redundant divisions of safety systems for test or maintenance purposes should be prevented. Safety system bypass was a key part of the cause of the Chernobyl accident. This issue called for the staff to recognize the lessons of Chernobyl in ongoing work to revise and improve Regulatory Guide 1.47.[150] Staff stated in the Supplement to NUREG-0933 published in 1989 that the scope of this issue included improved methods for indication of individual division bypass conditions and improved administrative controls over individual division bypasses. Completion of this issue would also resolve TMI Action Plan[48] Item I.D.3, "Safety System Status Monitoring." In revising Regulatory Guide 1.47,[150] the staff would: (a) evaluate the implications of bypassing safety systems; (b) recommend improved procedures and methods to prevent inadvertent bypassing of safety functions during test or maintenance; and (c) prepare revised Regulatory Guide 1.47 to reflect (a) and (b).

In pursuing this issue, the staff was expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue was considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated

November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.4: AVAILABILITY OF ENGINEERED SAFETY FEATURES

This item consists of three recommendations that are evaluated separately below.

ITEM CH1.4A: ENGINEERED SAFETY FEATURE AVAILABILITY

Description

ESF equipment needed to mitigate DBAs and transients currently have operability requirements in the TS to assure their availability for all modes of operation. In some instances, all of this equipment has not been evaluated in light of the need for its availability for plant shutdown modes. This issue called for the staff to evaluate and specify operability (availability) requirements for those ESF and support systems needed to mitigate DBAs and transients.

The issue will be addressed in the TSIP and is part of an overall program to ensure that the Owners' Groups and individual licensees specify the appropriate plant status modes for ESF equipment. In some of the older TS, mode requirements for operability may not be specified for other than the power operating mode. In the rewrite of the "Bases" sections of the TS, the reasons for LCOs will be included. Where the mode is currently absent or is inappropriately specified, the Bases will be clarified to identify required ESF equipment for each operational condition. However, ESF-required availability will only be addressed with respect to DBAs and transients and initial conditions (modes) currently analyzed in FSARs.

Reactor-vendor-based Owners' Groups will be permitted to remove those specifications in current STS that do not meet Commission criteria for what should be included in the TS. Requirements remaining in the TS will be rewritten and improved. Each rewritten and improved TS must have a Bases section that not only explains why a TS is needed, but also explains the plant conditions for which it is needed. This need will be evaluated for all of the operating modes of the plants.

Licensees will be encouraged to convert to the new STS and conduct similar upgrades for plant-unique specifications that meet the NRC criteria for the TS. These plant upgrades will be done on a voluntary basis. Those licensees participating will have appropriate ESF operability requirements specified for plant conditions where equipment could be needed for accident mitigative purposes. Upgraded plant-unique TS will also be evaluated. If significant ESF availability disparities are disclosed in this upgrade, they will be recommended for backfit on non-program participants' TS as the need arises.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the

Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.4B: TECHNICAL SPECIFICATIONS BASES

Description

Current TS Bases do not always provide a clear and comprehensive discussion linking specific requirements to the safety analysis assumptions they are derived from. This can result in operators not being as aware as possible of the safety significance of certain types of TS violations, an issue that may have had a counterpart at Chernobyl. It can also result in TS changes being proposed without adequate consideration of all the relevant safety issues. This issue called for the staff to develop an upgraded set of Bases for the STS to provide a clearer link between requirements and the safety analysis. The upgraded standard Bases will be made available to individual licensees for the purpose of adapting them to their plants as part of a voluntary industry-wide program to improve the TS.

It is planned that a separate set of upgraded standard Bases will be developed for each LWR design. The upgraded Bases will be developed as part of an ongoing joint NRC/Industry Technical Specifications Improvement Program (see SECY-86-310) that was initiated prior to the Chernobyl event. This is a program whereby the industry/utility owners' groups will completely rewrite the STS (including the Bases), making improvements in both format and content.

Once the new STS are developed, it is expected that most utilities will voluntarily elect to adopt them for their plants. Any decision to require an individual licensee to convert to the new STS will be made in accordance with the Backfit Rule (10 CFR 50.109). This issue is limited to the introduction of Chernobyl lessons into the staff's ongoing work and no separate work beyond that already started under the TSIP is planned. The Bases rewrite part of the Improvement Program will be comprehensive. A clear one-to-one relationship between TS requirements and the safety analysis will be documented in a carefully formatted Bases section for each TS. Separate Bases subsections will be written to address separate parts (i.e., LCOs, Action Statements, and Surveillance Requirements) of each plant's TS.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue is considered to be a licensing issue.

Conclusion

As reported in the Supplement to NUREG-0933 published in 1989, no incremental work attributable to Chernobyl lessons would be necessary to complete this issue. The only additional resources necessary would be those required to report progress against the Chernobyl Follow-up Research Plan and write a closeout report.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.4C: LOW POWER AND SHUTDOWN

Description

The Chernobyl event occurred when the unit was in a state of low power. In contrast, most regulatory attention and virtually all PRAs have focused on a state of full power operation. This issue called for the staff to perform an analysis of the core damage frequency and risk associated with a plant being in a state of low power or shutdown. The staff will examine the probabilistic risk from potential accidents initiated during shutdown and low power conditions at the Surry nuclear power plant.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue is considered to be a licensing issue.

Conclusion

As reported in the Supplement to NUREG-0933 published in 1989, the contract work on this task was being done as a part of the Accident Sequence Evaluation Program. Potential reactivity accident sequences that could originate at low or zero power were included in the scope of Item CH2.1A, "Reactivity Transients," the results of which might provide input to this issue.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.5: OPERATING STAFF ATTITUDES TOWARD SAFETY

Description

A significant aspect of the Chernobyl accident involved operator decisions and actions that reflected an apparent loss of the sense of vigilance toward safety and ultimately led to operators allowing operations outside the safe operating envelope. Some potential causes of this unacceptable attitude were: (1) pressure on the operators to complete a test during the reactor shutdown as the next opportunity would have been more than a year away; (2) test delay may have aggravated operator impatience and contributed to a "mindset" that led to imprudent safety actions; (3) operators, being so intent on establishing acceptable power level for the test, may have ignored the unstable state of the reactor; and (4) a clear failure to appreciate the basic reactor physics of the RBMK reactor. The accident raised the question whether licensed operators, senior operators, and other staff at nuclear power plants in the U.S. have and maintain an acceptable level of vigilance toward safety when operating commercial nuclear power plants.

In pursuing this issue, the staff increased its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Thus, the issue was considered to be a licensing issue.

Conclusion

The staff believes that safeguards against unacceptable operator and plant personnel attitudes toward safety are adequate. This conclusion is based on the significant increase in the quality of training, industry initiatives in accrediting training programs, and regulatory and industry oversight inspections. Thus, this Licensing Issue has been resolved.

ITEM CH1.6: MANAGEMENT SYSTEMS

This item consists of one recommendation that is evaluated below.

ITEM CH1.6A: ASSESSMENT OF NRC REQUIREMENTS ON MANAGEMENT

Description

Management oversight at all levels must be effective to ensure that tests, maintenance, and operations are conducted safely and that NRC requirements are enforced. The NRC is developing improved methods of monitoring licensee management performance to give early warning of management problems and to initiate enforcement mechanisms. It is also important to ensure that the monitoring and evaluation of management systems consider management capability to handle emergencies and the immediate effects of an accident. Issues of importance include management measures to ensure the availability of personnel capable of handling emergencies, planning for the operation of plant controls and systems with severe core damage, and plant staff training for operation under severe emergency conditions. At the same time, it is important that NRC-imposed requirements on management be reasonable and without excessive burdens that could divert from critical responsibilities. Management failure to recognize and respond appropriately to hazardous conditions was a major factor in the Chernobyl accident. This issue called for the staff to ensure that NRC research programs involving the review or evaluation of utility management include the management issues arising from the Chernobyl event, with particular attention to matters important to safety and to avoidance of excessive burdens that could divert that attention.

The staff will coordinate activities of the following research programs that focus on the performance of utility management to ensure that the concerns of this issue are being addressed cohesively: (1) Management/Organization Influence on Human Error Rates; and (2) Programmatic Performance Indicators. Activities of any new research programs in this area, e.g., Severe Accident Management, will be coordinated for the same purpose. The staff will also coordinate the development of the following evaluation techniques:

(a) Management capability to handle severe accidents of the Chernobyl scale; (b) Management measures requiring the availability of personnel capable of handling emergencies of the type experienced at Chernobyl; (c) Management programs for training personnel to handle emergencies; and (d) Management plans for the operation of plant controls and systems to cope with severe core damage. Coordination will be extended to the following identified user needs as these needs are translated into research programs: (1) Operator Performance Under Stress of Emergency Operations; and (2) Severe Accident Management.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH1.7: ACCIDENT MANGEMENT

This item consists of one recommendation that is evaluated below.

ITEM CH1.7A: ACCIDENT MANAGEMENT

Description

The consideration of severe accidents in current symptom-based procedures typically does not go beyond the area of inadequate core cooling. This issue called for the staff to coordinate NRC research activities and programs dealing with severe accident management to ensure the appropriate incorporation of insights gained from the Chernobyl event. This may involve the review of severe accident management programs that may be implemented at existing nuclear power plants. The staff will: (a) assist in scoping the training, organization and habitability elements of new research programs addressing severe accident management to incorporate the Chernobyl lessons learned; (b) review ongoing NRC severe accident management programs and recommend modifications as needed to include the insights gained from the Chernobyl event; and (c) participate in NRC reviews of individual plant severe accident management programs and determine the extent to which these programs have taken advantage of the insights gained from the Chernobyl event.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

150. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Atomic Energy Commission, May 1973. [7907100191]

1174. NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," U.S. Nuclear Regulatory Commission, (Vols. I and II) April 1989.

1176. AEOD/S801, "Significant Events that Involved Procedures," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 1988. [8907310351, 8906090032]

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

TASK CH2: DESIGN

The Chernobyl Unit 4 accident was a prompt critical reactivity excursion that occurred when the operators reduced power to well below the permissible safe operating level and, at the same time, neglected to follow low power operating procedures. Unit 4 shared a site with Units 1, 2, and 3 and was contiguous with Unit 3 with which it also shared some common elements. All three of the other units were exposed to some danger from the accident. Fires aggravated the accident and complicated its management and consequences. In this task, outlined in Chapter 2 of NUREG-1251,[1174] the staff will compare the design features of U.S. reactors with those of the Chernobyl 4 reactor in looking for possible regulatory changes implicit in the accident.

ITEM CH2.1: REACTIVITY ACCIDENTS

This item consists of one recommendation that is evaluated below.

ITEM CH2.1A: REACTIVITY TRANSIENTS

Description

In light of Chernobyl, it is necessary to examine some of the multiple-failure reactivity transients using PRA tools to reconfirm previous judgments. This item called for the staff to perform a study to estimate probability levels of certain reactivity transients. If any events appear to fall within the probability levels of NRC guidelines and involve a significant potential for extensive core damage, they might become a basis for changing design or operational limits. The study will include both probabilistic analyses to estimate the frequency of an event and deterministic analyses to assess the potential consequences. The events of interest are those in which there is a relatively large reactivity insertion and/or the response of the shutdown system may be inadequate. Identified events of interest are:

BWRs

- Multiple rod drop
- Control rod ejection
- Overpressurization with limited relief
- Boron dilution during anticipated transient without scram (ATWS)
- ATWS without recirculation pump trip
- Multiple rod bank withdrawal
- Reactivity events with more than one rod stuck out

PWRs

- Multiple rod blank withdrawal ATWS
- Multiple rod ejection (low power)
- Injection of cold, unborated emergency cooling water
- Injection of cold, unborated water due to SGTR
- Unlimited boron dilution

- Rod withdrawal, heatup or depressurization from low temperature with positive moderator temperature coefficient
- ATWS with less negative moderator temperature coefficient
- Reactivity events with more than one rod stuck out

In addressing this issue, the staff will focus attention on sequences that might involve a positive void coefficient or moderator temperature coefficient, that might arise in connection with deliberate bypassing or disabling of any safety feature, and whose causes include human error (commission, omission, or misjudgment).

The six parts of this issue are as follows:

I. <u>Establishment of Criteria:</u> Criteria will be established to judge whether a particular sequence needs further examination by the NRC.

II. <u>Selection of Events:</u> Sequence of event trees will be developed for the events identified above and critical sequence paths will be determined for different modes of reactor operation in light of positive moderator temperature coefficient, deliberate bypassing or disabling of any safety feature and human errors including commission, omission, and misjudgments. One typical Westinghouse PWR (Byron) and one typical BWR (Peach Bottom) were chosen to be analyzed. If certain sequences in certain events are important, analyses will be extended to other types of plants.

III. <u>Probabilistic Quantification of Events:</u> The accident sequences that emerge from Part II will be quantified to establish those that meet criteria in Part I above. The quantification process will involve a detailed search of various data bases to obtain failure rates and event probabilities. If the data base is not available, such as in the case of human errors, conservative assumptions will be made.

IV. <u>Physical Assessment of Events:</u> For each sequence of events for which the frequency of occurrence is either unknown or expected to be significant according to the criteria of Part I, a deterministic analysis will be made. Key parameters will be determined and their limiting values quantified. The quantification will be done primarily by using results of analyses which have already been performed for other purposes.

V. <u>Preparation of Report:</u> A draft report will be prepared integrating the above described tasks.

VI. <u>Final Report:</u> A final report will be prepared after comments.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated

November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH2.2: ACCIDENTS AT LOW POWER AND AT ZERO POWER

Description

One of the unique aspects of the Chernobyl accident is that it occurred at relatively low power (<7%). This has caused some concern because low power operation is generally considered to be a safer condition than high or full power operation. The principal effect of low power on the Chernobyl accident was related to nuclear/thermohydraulic stability and reactivity insertion. These effects were addressed in Item CH2.1. Another important aspect of low power or zero power operation is the availability of safety systems. Items CH1.3 and CH1.4 specifically address the subjects of bypassing and availability of safety systems. Different safety systems may be used to provide protection for low power and shutdown (zero power) events than are used for high power events. TS prescribe the conditions for bypassing and activating the various systems and their completeness is also addressed in Items CH1.3 and CH1.4.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

Accident initiators at low power are covered in Item CH1.4 which is to be coordinated with the Severe Accident Program. The results of Item CH1.4 will be made available to the industry to help develop TS improvements if necessary.

ITEM CH2.3: MULTIPLE-UNIT PROTECTION

The radioactive gas and smoke released during the accident at Chernobyl Unit 4 spread to the other three operating units at the site. The airborne radioactive material was transported to the other units through a shared ventilation system as well as by way of general atmospheric dispersion paths. This raises the question of how accidents at one unit of a multi-unit site affect the remaining units and additional questions of how these effects may be compounded when structures, systems, and components are shared between units. This item consists of four recommendations that are evaluated separately below.

ITEM CH2.3A: CONTROL ROOM HABITABILITY

Description

The objective of this issue is to estimate what effects an accident at one unit of a multi-unit site could have upon the ability of site personnel to maintain the remaining units in a safe condition, to identify potential new requirements that would decrease those effects, and to assess the safety advantages of such requirements in relation to the disadvantages of their imposition. Although identified as a multi-unit issue, the staff's work should include site emergencies such as fires and other potential causes of widespread damage that might not be directly related to a particular unit. By including control room habitability challenges not initiated by a reactor accident, single unit sites would also be included.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

All efforts to address this issue are included in the plans for the resolution of Issue 83, "Control Room Habitability." Included in these plans is a survey of a sample of U.S. control rooms at diverse plants and sites and an assessment of the capabilities of these control rooms and their habitability systems to meet GDC 5 and 19. In the event of deficiencies in the assessed capabilities, the costs and benefits of backfits needed to achieve those capabilities are to be assessed and, where justified, requirements specified.

ITEM CH2.3B: CONTAMINATION OUTSIDE CONTROL ROOM

Description

The objective of this issue is to identify all plant areas to which human access would be necessary to either manage an accident at an affected unit or to maintain other units at a multi-unit site, to assess the dose consequences to personnel performing needed tasks within those areas, and to identify any potential measures for further reducing those consequences which could be justified by virtue of improved risk.

The necessary information to perform the work required by this issue includes identification of risk-dominant accidents and their corresponding accident management plans. For the identified accidents and the associated plant areas to which access is needed, generic estimates of contamination of those areas, in combination with generic measures of radiation shine from adjacent equipment and from other units, need to be developed.

The identification of plant areas to which access is required occurred during resolution of TMI Action Plan[48] Item II.B.2. It will be confirmed that these plant identifications are consistent with the accident management considerations being proposed in conjunction with the IPE. This work is incorporated in existing efforts in accident management research.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

This item consisted of review and coordination to assure that Chernobyl lessons were taken into account in the Accident Management Research Plan. The results of this issue would constitute an input to the Accident Management Research efforts.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH2.3C: SMOKE CONTROL

Description

This issue called for the staff to assess the risk significance of smoke propagation from one unit to an adjacent unit and to address the question of whether additional protection/requirements should be developed. The staff will use fire risk assessments from four LWRs to assess the risk significance of smoke propagation. Based upon the results, the need for further work will be determined. This issue could affect existing and future plants.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH2.3D: SHARED SHUTDOWN SYSTEMS

Description

This issue called for the staff to determine whether sharing of systems required for safe shutdown among units at a multi-unit site should be prohibited and, if not, to what restrictions such sharing should be subjected. The staff is to determine requirements for shared systems and prepare guidance on the use of shared systems as part of the severe accident policy implementation. It is anticipated that only future plants will be affected by this issue.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH2.4: FIRE PROTECTION

This item consists of one recommendation that is evaluated below.

ITEM CH2.4A: FIREFIGHTING WITH RADIATION PRESENT

Description

This issue called for the staff to determine: (1) whether there is a significant risk that radiation released during a fire or from the initiating event could limit firefighting capability; and (2) what additional measures, if any, such risk might necessitate. The staff will use existing representative fire risk studies from four LWRs to estimate risk. This issue could affect existing and future plants.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Rev. 1) August 1980.

1174. NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," U.S. Nuclear Regulatory Commission, (Vols. I and II) April 1989.

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, "Summary of Activities Related to Generic Issues Program," July 26, 2011. [ML111590814]

TASK CH3: CONTAINMENT

The Chernobyl accident, with its absence of effective containment, has focused attention on the strengths and performance limits of the substantial containments for U.S. LWRs. It has led to added recognition of the significance of ongoing work on the issue of whether U.S. containments that were built using criteria based on DBAs have adequate margins available to prevent the release of large quantities of fission products during severe accidents. Challenges include phenomena such as increased pressures from an uncontrolled hydrogen combustion or release of large quantities of noncondensible gases from core-concrete interactions. Venting the containment in case of certain severe accidents could be an effective way to preserve the long-term containment functional integrity and reduce the uncontrolled release of radioactive material. This task, outlined in Chapter 3 of NUREG-1251,[1174] summarizes the activities already in place in the areas of containment integrity and containment venting.

ITEM CH3.1: CONTAINMENT PERFORMANCE DURING SEVERE ACCIDENTS

This item consists of one recommendation that is evaluated below.

ITEM CH3.1A: CONTAINMENT PERFORMANCE

Description

This issue called for the staff to determine whether the Chernobyl containment failure indicates that changes in U.S. containment or reactor design and operation requirements are warranted. In addressing this issue, the staff is expected to reflect Chernobyl experience, where relevant, in containment reviews under the Commission's Severe Accident Policy.

An existing set of tasks relating to adequate containment performance was underway in the U.S. before the Chernobyl accident. These tasks (IPE, the development of accident management strategies, containment performance, and NUREG-1150)[1081] are related to determining whether the existing design and operation of U.S. commercial reactors provide an adequate level of safety or whether changes in regulatory guidance are required. The Chernobyl accident adds to the information base only indirectly because of differences in reactor types and containment (or confinement) approaches.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue considered is to be a licensing issue.

Conclusion

Efforts to address this issue were underway, as noted in SECY-87-297. No separate projects or assessments were envisaged.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH3.2: FILTERED VENTING

This item consists of one recommendation that is evaluated below.

ITEM CH3:2A: FILTERED VENTING

The issue called for the staff to determine whether U.S. containments should be backfitted with filtered vents to mitigate the consequences of severe accidents as is being proposed and implemented in Europe. The Chernobyl accident heightened interest in this issue, though the issue itself has no specific Chernobyl counterpart. The purpose of this issue is to develop information to be used in assessing filtered vents proposed for U.S. reactors and to advise the Commission on whether such systems should be required for specific categories of U.S. reactors. The staff will assess the filtered venting technology emerging from European research and applications for potential U.S. reactor severe accident improvements. This work is a non-distinguishable part of the development of accident management strategies and containment performance assessments.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue considered is to be a licensing issue.

Conclusion

As reported in the Supplement to NUREG-0933 published in 1989, venting was being studied by INEL under staff contracts. This study required an assessment of European research and applications and keeping abreast of relevant literature and participation in international evaluation activities. One such activity was the Nuclear Energy Senior Group of Experts on Severe Accidents meeting on Filtered Containment Venting Systems held in May 1988 in Paris and the preparation of a "white paper" on the technology and related issues. No separate projects or assessments arising from Chernobyl were envisaged.

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

References

1081. NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

1174. NUREG-1251, “Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States,” U.S. Nuclear Regulatory Commission, (Vols. I and II) April 1989.

1858. Management Directive 6.4, “Generic Issues Program,” U.S. Nuclear Regulatory Commission, November 17, 2009.

1967. SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” July 26, 2011. [ML111590814]

TASK CH4: EMERGENCY PLANNING

A number of facts about the Chernobyl accident have some bearing on emergency planning and preparedness around U.S. commercial nuclear power plants. This task, outlined in Chapter 4 of NUREG-1251,[1174] called for the staff to examine the implications of the accident and the Soviet response for four aspects of U.S. emergency planning: (1) size of the emergency planning zone (EPZ); (2) medical services; (3) ingestion pathway measures; and (4) decontamination and relocation.

ITEM CH4.1: SIZE OF THE EMERGENCY PLANNING ZONES

Description

The Chernobyl accident focused attention on the adequacy of the size of EPZs around U.S. commercial nuclear power plants. The Soviets evacuated a total of about 135,000 people as well as considerable farm livestock from Pripyat, Chernobyl, and other towns and villages within 18 miles of the Chernobyl power plant. This evacuation appears to have taken place in several stages, beginning for the approximately 45,000 residents of Pripyat about 36 hours after the initial release and extending over several days to a week. The whole-body radiation dose to the majority of individuals did not exceed 25 rem, although about 24,000 persons in the most severely contaminated areas are estimated to have been exposed to whole-body doses in the range of 35 to 55 rem. The population of Pripyat was initially sheltered as a protective measure and then evacuated when radiation readings increased. In addition to radiation considerations, logistics and contamination control influenced the timing of the evacuation. Despite an apparent lack of site-specific planning, the Soviets mounted a large and generally effective ad hoc response making use of some aspects of civil defense planning. The high initial plume contributed to relatively low initial dose rates in the immediate vicinity. In addition, efforts by the Soviets to prevent rainfall in the immediate vicinity (by cloud seeding other areas) and the spraying of a chemical polymer on evacuation routes to minimize resuspension of deposited activity were also beneficial. The Soviets took ingestion pathway protective measures within the 18-mile zone and well beyond. Ingestion pathway protective measures were also taken in several Soviet bloc countries, in Scandinavia, and in Eastern and Western Europe.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

The Chernobyl accident and the Soviet response did not reveal any apparent deficiency in U.S. plans and preparedness, including the 10-mile plume exposure pathway EPZ size and and the 50-mile ingestion exposure pathway EPZ size. These zones provide an adequate basis to plan and carry out the full range of protective actions for the population within these zones as well as beyond them, if the need should arise. Any changes in EPZ sizes should be based on revised insights coming from current U.S. research on severe accident releases. No recommendation resulted from this item which was dropped from further consideration.

ITEM CH4.2: MEDICAL SERVICES

Description

At Chernobyl, KI was distributed to school children within about 6 hours of the accident and to the entire population of Pripyat the morning of the following day; ultimately, it was given to the population in the 18-mile zone and other areas. The Soviets reported no serious adverse reactions to KI. Polish authorities also distributed KI to the population in parts of eastern Poland. This issue called for the staff to review the adequacy of the U.S. Government's policy on KI and the adequacy of medical services around U.S. nuclear power plants.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

The apparently successful use of KI by the Soviets did not alter the validity of U.S. Government policy that pre-distributing or stockpiling KI for use by the general public should not be required; rather, this decision should be made by individual States and by local authorities. Further, the staff concluded that the present arrangements and future plans for medical services around U.S. commercial nuclear power plants are adequate. The national capability is both substantial and growing. Also, the international offers of medical support to the Soviet Union following the Chernobyl accident demonstrate that the U.S. regional and national medical response can be augmented, if necessary by a response from the international medical community. No recommendation resulted from this item which was dropped from further consideration.

ITEM CH4.3: INGESTION PATHWAY MEASURES

This item consists of one recommendation that is evaluated below.

ITEM CH4.3A: INGESTION PATHWAY PROTECTIVE MEASURES

Description

After the Chernobyl accident, human and animal food chains in the Soviet Union and other European countries were contaminated to varying degrees. The Soviet and other affected governmental authorities took measures, both short-term and long-term, to protect the public from receiving unacceptably high levels of radiation through consumption of contaminated food. The contamination level findings and the experience with the Soviet and other European control measures could provide important extensions of the data base for planning of protective measures in the U.S. This issue called for the staff to participate with FEMA and other Federal and appropriate international agencies in planning and eventual execution of efforts to obtain available information on the Soviet and other European post-Chernobyl ingestion pathway contamination and control measures experience and analyze that information in relation to U.S. understanding of the issue.

The work is expected to be done primarily under FEMA's coordination together with other appropriate Federal agencies, such as FDA and EPA, and international agencies such as IAEA. The NRC will participate in this work to assure adequate representation of NRC's interest in the effort and to obtain the information needed for NRC's purposes. The information to be sought is expected to encompass contamination level findings for various human and animal foodstuffs, as well as water bodies, including variation with time and place, and the nature, timing, effectiveness, and problems of various protective measures taken by the affected countries. Future analyses are expected to relate findings to U.S. source term research results. The work of CY 1988 is expected to be devoted primarily to establishment of interagency and international contracts and arrangements and development of a research plan, in cooperation with FEMA and other agencies. The plan is expected to encompass both near-term work, focusing on the short-term experience, and long-term plans for a number of future years, for lessons of the long-term experience.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, “Summary of Activities Related to Generic Issues Program,” dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, “Generic Issues Program,” dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH4.4: DECONTAMINATION AND RELOCATION

This item consists of two recommendations that are evaluated separately below.

ITEM CH4.4A: DECONTAMINATION

Description

The practicality and effectiveness of measures to decontaminate structures, land, etc. after a major accident can be a significant factor in evaluation of accident consequences as well as in formulation of plans and approaches for post-accident decontamination. The experience with post-Chernobyl decontamination in the Soviet Union could provide important extensions of the data base. This issue called for the staff to participate with FEMA and other Federal and international agencies in planning and eventual execution of efforts to obtain available information on the Soviet post-Chernobyl decontamination experience and analyze that information in relation to U.S. understanding of the issue.

The work is expected to be done primarily under FEMA's coordination, together with other appropriate federal agencies such as EPA and FDA and international agencies such as IAEA.

The NRC will participate in this work to assure adequate representation of NRC's interest in the effort and to obtain the information needed for NRC's purposes. The information to be sought is expected to encompass methods, timing, and effectiveness of decontamination of various areas and objects. Future analyses are expected to relate findings to U.S. source term research results. The work in CY 1988 is expected to be devoted primarily to establishment of interagency and international contacts and arrangements and development of a research plan, in cooperation with FEMA and other agencies, in connection with acquisition and analysis of Soviet information that may become available over the next several years.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH4.4B: RELOCATION

Description

Notwithstanding cultural and socioeconomic differences, the Soviet experience in connection with post-accident evacuation and relocation of the population of contaminated towns and villages near the Chernobyl reactor may well offer valuable lessons for U.S. emergency planning. This issue called for the staff to participate, with FEMA and other appropriate Federal and international agencies, in developing plans and arrangements for learning about and from the Soviet post-Chernobyl relocation experience.

Plans and interagency and international arrangements will be developed, under FEMA coordination, together with other Federal agencies and international bodies such as IAEA. Logistical, socioeconomic, health, and psychological considerations are expected to be included in the information to be sought.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

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TASK CH5: SEVERE ACCIDENT PHENOMENA

The highly energetic reactivity excursion accident at Chernobyl mechanically disrupted the core, rapidly vaporized the water coolant with which the fragmented fuel came into contact, and generated combustible hydrogen by chemical reaction of core materials (notably zirconium) and water at the high temperatures reached in the accident. Because of basic design differences between the RBMK reactor of Chernobyl and U.S. LWRs, the specific accident mechanisms involved at Chernobyl have no exact parallel in U.S. reactors. However, this task, outlined in Chapter 5 of NUREG-1252,[1174] called for the staff to assess Chernobyl phenomena for analogous implications of radionuclide releases, steam explosions, and combustible gas generation and deflagration control in U.S. reactors.

ITEM CH5.1: SOURCE TERM

This item consists of two recommendations that are evaluated separately below.

ITEM CH5.1A: MECHANICAL DISPERSAL IN FISSION PRODUCT RELEASE

Description

The initial release of fission products that occurred at Chernobyl was the result of mechanical dispersion. Such a mechanism is possible in LWRs within the containment during energetic events such as high pressure melt ejection, steam explosions, and hydrogen combustion. Although such events are being studied with regard to their likelihood of occurrence and their consequences, associated mechanical releases of fission products have not been quantified in current source term models and the study of such releases has only just begun to receive attention. Because some of these phenomena appear to have played a dominant role in the releases at Chernobyl, it is important to understand these phenomena more completely. This issue called for the staff to introduce the Chernoby lessons into ongoing work to improve the understanding of mechanical dispersal phenomena and to improve the modeling in NRC source term assessment codes.

Current research on mechanical dispersion is being performed in three specific areas: direct containment heating (or high pressure melt ejection), steam explosions, and hydrogen combustion. For direct containment heating, the scope of current research is to develop a capability to analyze the consequences of this phenomenon. This can be accomplished by generating an experimental data base and, by developing an analytical model based on this data base which will be subsequently incorporated in an integrated code for containment analyses. In the area of hydrogen combustion, present work includes a scoping study on mechanisms of aerosol re-suspension and volatilization during hydrogen combustions. Specifically, experiments are being conducted to investigate the re-suspension of aerosols (radioactive or otherwise) that have been previously deposited on containment surfaces, by mechanical or thermal processes during the occurrence of hydrogen combustion, and to investigate the volatilization and expulsion of airborne aerosols in the containment by similar processes. The new information will subsequently be incorporated into the lumped parameter code HECTR and the finite difference code HMS-BURN for consequence analyses.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH5.1B: STRIPPING IN FISSION PRODUCT RELEASE

Description

The late enhanced release of fission products during the Chernobyl accident may be attributable to the chemical and/or thermal stripping of UO_2 fuel. Such mechanisms have been observed in in-pile and out-of-pile experiments when UO_2 fuel rods were exposed to steam or high temperatures and other severe degraded core conditions. During the process of thermal stripping, for example, fission products were released in proportion to the amount of UO_2 vaporized. The rate of fission product release is thus controlled by UO_2 vaporization.

Fission product release by chemical and thermal stripping mechanisms is not modeled in current severe accident source term codes. The Chernobyl accident has demonstrated that such mechanisms can be important in fission product release under some conditions. This issue called for the staff to introduce Chernobyl lessons into the continuing research on chemical and thermal stripping and to obtain sufficient data for model development and assessment.

The scope of present research on UO_2 stripping is to complete ongoing experiments investigating thermal stripping mechanisms, to collect and review experimental data on chemical stripping mechanisms from Severe Fuel Damage Program participants, and to apply both the thermal stripping and chemical stripping data to improve present fission product release codes. For chemical stripping, the present experimental program may have to be expanded to study UO_2 stripping by air oxidation. This recommendation involves coordination to assure that the ongoing work adequately reflects the Chernobyl lessons.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH5.2: STEAM EXPLOSIONS

This item consists of one recommendation that is evaluated below.

ITEM CH5.2A: STEAM EXPLOSIONS

Description

No specific research is currently underway or planned on reactivity insertion accident (RIA) prompt-burst steam explosions with fuel-vapor-driven fragmentation and mixing of the molten fuel and water that are relevant to the Chernobyl accident. Such work is currently not believed to be necessary, subject to confirmation in the light of results of the Chernobyl follow-up reactivity transient study (Item 2.1A).

The vapor-driven fragmentation and mixing of the interspersed fuel and coolant in prompt-burst power excursions in the Chernobyl accident has been strongly contrasted in the past to the pouring mode of contact found in the slow meltdown situations relevant to current U.S. commercial reactors. Hence the Chernobyl accident has little relevance to the staff's current treatment of steam explosions and alpha-mode containment failure. This issue called for the staff to characterize RIA steam explosions.

Current steam explosion research consists primarily of developing and assessing the semi-mechanistic Integrated Fuel Coolant Interaction (IFCI) computer model, which includes hydrogen generation, for integration into an in-vessel melt progression code. IFCI provides a mechanistic treatment of both the pre-explosion mixing phase and the explosion phase (if conditions permit), but IFCI does require a parametric input trigger for the explosion. Work is also continuing on using existing experimental data for modeling the non-explosive mixing phase of the interaction.

If further work for U.S. reactors on RIA steam explosions is found to be needed, this would be performed as part of an overall investigation of RIAs and it is in this context that the specific work scope would be planned. Currently work is underway to assess the effect of in-vessel steam explosions on in-vessel core melt progression in light-water reactor accidents.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, this issue is considered to be a licensing issue.

Conclusion

As a part of the improvements to NUREG-0933, the NRC staff clarified in SECY-11-0101, "Summary of Activities Related to Generic Issues Program," dated July 26, 2011,[1967] that the Generic Issues Program will not pursue any further actions toward resolution of licensing and regulatory impact issues. Because licensing and regulatory impact issues are not safety issues by the classification guidance in the legacy Generic Issues Program, these issues do not meet at least one of the Generic Issues Program screening criteria and do not warrant further processing in accordance with Management Directive 6.4, "Generic Issues Program," dated November 17, 2009.[1858] Therefore, this issue will not be pursued any further in the Generic Issues Program.

ITEM CH5.3: COMBUSTIBLE GAS

Description

The Soviet RBMK design utilizes large amounts of zirconium and graphite in the reactor core, both of which may oxidize under certain conditions resulting in the generation of large quantities of combustible gases, principally hydrogen and carbon monoxide. The generation of large quantities of combustible gases was not apparently considered as part of the Soviet containment design. The Chernobyl accident produced reactor core conditions that may have led to the generation of large quantities of combustible gases which, in turn, may have influenced the evolution and consequences of the accident.

The need to deal with the generation of combustible gas, principally hydrogen, as a consequence of reactor accidents has been recognized in the U.S. since the early days of LWRs. The burning and/or detonation of combustible gases are of concern in reactor safety for several reasons. First, a large enough energy release might threaten the integrity of the containment. Second, even if the containment survived, important safety equipment might be irreparably damaged, thus increasing the severity of the accident. Furthermore, since significant amounts of hydrogen can be generated early in the evolution of a severe reactor accident (i.e., before the reactor vessel fails), combustion can result in containment failure before expulsion of the molten core, leading to the largest radioactivity releases to the environs.

Conclusion

In summary, although the conditions that existed during the Chernobyl accident may have caused large amounts of combustible gases to generate, it cannot be concluded from the available data that these gases were generated by some new or different mechanisms or produced consequences not previously investigated as part of severe-accident analyses for U.S. reactors. It is difficult to apply observations from the Chernobyl accident to U.S. plants because of significant design differences between the RBMK and nuclear power reactors in the United States; furthermore, the NRC staff still lacks detailed accident data. Considering the preliminary evaluation, it does not appear that any additional work is warranted solely on the basis of the Chernobyl event. The staff concludes that its current and proposed research program on combustible gas phenomena in conjunction with the study of severe accidents would be adequate for addressing this issue in U.S. reactors.

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APPENDIX B

APPLICABILITY OF NUREG-0933 ISSUES TO OPERATING AND FUTURE REACTOR PLANTS

This appendix contains a list of those generic safety issues (GSIs) that are applicable to operating and future reactor plants, including issues that have been resolved with requirements (e.g., I, NOTE 3(a)) and issues that are in progress for resolution. The priority designations for all issues are consistent with those listed in Table II of the Introduction to NUREG-0933. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(21), applications for design certification must contain proposed technical resolutions of those unresolved safety issues, high- and medium-priority generic safety issues, which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design. Similarly, in accordance with 10 CFR 52.79(a)(20), applications for combined licenses must contain proposed technical resolutions of those unresolved safety issues, high- and medium-priority generic safety issues, which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design.

In Management Directive (MD) 6.4, “Generic Issues Program,” first issued July 21, 1999, the U.S. Nuclear Regulatory Commission replaced prioritization of generic issues (GIs) with the screening process, in which staff determines to either establish the proposed issue as a bone fide GI or reject the issue from the program. For the purposes of 10 CFR 52.47(a)(21) and 10 CFR 52.79(a)(20), any GI established by the MD 6.4 screening process is considered equivalent to a high-Priority GI.

<u>Legend</u>

ACTIVE	Work on the issue continues in accordance NRC Management Directive 6.4
B&W	Babcock & Wilcox Company
CE	Combustion Engineering Company
GE	General Electric Company
I	Resolved Three Mile Island (TMI) Action Plan item with implementation of resolution mandated by NUREG-0737
NOTE 3(a)	Resolution resulted new regulatory products (rule, regulatory guide, SRP change, or equivalent)
ROI	Regulatory office implementation: A formal GI for which Office of Nuclear Regulatory Research actions of safety/risk assessment or regulatory assessment are complete and remaining actions reside with program offices (e.g., regulatory compliance, Reactor Oversight Process, rulemaking, further research, coordination with industry initiatives)
MEDIUM	Medium safety priority
MPA	Multiplant action
NA	Not applicable
TBD	To be determined
USI	Unresolved safety issue
<u>W</u>	Westinghouse Electric Corporation

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	I	All	All	F-01	09/13/79	09/27/79
I.A.1.2	Shift Supervisor Administrative Duties	I	All	All	-	09/13/79	09/27/79
I.A.1.3	Shift Manning	I	All	All	F-02	07/31/80	06/26/80
I.A.1.4	Long-Term Upgrading	NOTE 3(a)	All	All	-	04/28/83	04/28/83
I.A.2	Training and Qualifications of Operating Personnel						
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications						
I.A.2.1(1)	Qualifications—Experience	I	All	All	F-03	03/28/80	03/28/80
I.A.2.1(2)	Training	I	All	All	F-03	03/28/80	03/28/80
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	I	All	All	F-03	03/28/80	03/28/80
I.A.2.3	Administration of Training Programs	I	All	All	-	03/28/80	03/28/80
I.A.2.6	Long-Term Upgrading of Training and Qualifications						
I.A.2.6(1)	Revise Regulatory Guide 1.8	NOTE 3(a)	All	All	-	TBD	05/87
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	I	All	All	-	03/28/80	03/28/80
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement						
I.A.4.1(2)	Interim Changes in Training Simulators	NOTE 3(a)	All	All	-	04/81	03/28/81
I.A.4.2	Long-Term Training Simulator Upgrade						
I.A.4.2(1)	Research on Training Simulators	NOTE 3(a)	All	All	-	04/87	04/87
I.A.4.2(2)	Upgrade Training Simulator Standards	NOTE 3(a)	All	All	-	04/81	04/81
I.A.4.2(3)	Regulatory Guide on Training Simulators	NOTE 3(a)	All	All	-	04/81	04/81

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
I.A.4.2(4)	Review Simulators for Conformance to Criteria	NOTE 3(a)	All	All	-	03/25/87	03/25/87
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision						
I.C.1(1)	Small-Break LOCAs	I	All	All	-	09/13/79	09/13/79
I.C.1(2)	Inadequate Core Cooling	I	All	All	F-04	09/13/79	09/13/79
I.C.1(3)	Transients and Accidents	I	All	All	F-05	09/13/79	09/27/79
I.C.2	Shift and Relief Turnover Procedures	I	All	All	-	09/13/79	09/27/79
I.C.3	Shift Supervisor Responsibilities	I	All	All	-	09/13/79	09/27/79
I.C.4	Control Room Access	I	All	All	-	09/13/79	09/27/79
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	I	All	All	F-06	05/07/80	06/26/80
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	I	All	All	F-07	10/31/80	10/31/80
I.C.7	NSSS Vendor Review of Procedures	I	All	All	-	NA	06/26/80
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	I	All	All	-	NA	06/26/80
I.C.9	Long-Term Program Plan for Upgrading of Procedures	NOTE 3(a)	All	All	-	09/13/79	06/85
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	I	All	All	F-08	06/26/80	06/26/80
I.D.2	Plant Safety Parameter Display Console	I	All	All	F-09	06/26/80	06/26/80
I.D.5	Improved Control Room Instrumentation Research						
I.D.5(2)	Plant Status and Post-Accident Monitoring	NOTE 3(a)	All	All	-	NA	12/80
I.F	QUALITY ASSURANCE						
I.F.2	Develop More Detailed QA Criteria						
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	NOTE 3(a)	All	All	-	NA	07/81

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	NOTE 3(a)	All	All	-	NA	07/81
I.F.2(6)	Increase the Size of Licensees' QA Staff	NOTE 3(a)	All	All	-	NA	07/81
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	NOTE 3(a)	All	All	-	NA	07/81
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	I	All	All	-	NA	06/26/80
I.G.2	Scope of Test Program	NOTE 3(a)	All	All	-	NA	07/81
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	I	All	All	F-10	09/13/79	09/27/79
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	I	All	All	F-11	09/13/79	09/27/79
II.B.3	Post-Accident Sampling	I	All	All	F-12	09/13/79	09/27/79
II.B.4	Training for Mitigating Core Damage	I	All	All	F-13	03/28/80	03/28/80
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	NOTE 3(a)	All	All	-	TBD	NA
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	NOTE 3(a)	All	All	-	TBD	01/25/85
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	I	All	All	F-14	09/13/79	09/27/79
II.D.3	Relief and Safety Valve Position Indication	I	All	All	-	07/21/79	09/27/79
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	I	NA	All	F15	03/10/80	03/10/80

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	I	NA	All	F-16, F-17	09/13/79	09/27/79
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	NOTE 3(a)	All	All	-	NA	07/81
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	I	NA	All	-	09/13/79	09/27/79
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	I	All	All	F-18	09/13/79	09/27/79
II.E.4.2	Isolation Dependability	I	All	All	F-19	09/13/79	09/27/79
II.E.4.4	Purging						
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	NOTE 3(a)	All	All	-	11/28/78	NA
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	NOTE 3(a)	All	All	-	10/22/79	NA
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	NOTE 3(a)	All	All	-	09/27/79	NA
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	NOTE 3(a)	NA	B&W	-		
II.E.5.2	B&W Reactor Transient Response Task Force	NOTE 3(a)	NA	B&W	-		
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	NOTE 3(a)	All	All	-	06/89	06/89
II.F	INSTRUMENTATION AND CONTROLS						
II.F.1	Additional Accident Monitoring Instrumentation	I	All	All	F-20, F-21 F-22, F-23 F-24, F-25	09/13/79	09/27/79
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	I	All	All	F-26	070/2/79	09/27/79
II.F.3	Instruments for Monitoring Accident Conditions	NOTE 3(a)	All	All	-	NA	12/80

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	I	NA	All	-	09/13/79	09/27/79
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	NOTE 3(a)	All	All	-	07/31/91	07/31/91
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins						
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	NOTE 3(a)	All	All	-	03/31/80	NA
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	NOTE 3(a)	NA	B&W	-	03/31/80	NA
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	NOTE 3(a)	NA	All	-	03/31/80	NA
II.K.1(4)	Review Operating Procedures and Training Instructions	NOTE 3(a)	All	All	-	03/31/80	NA
II.K.1(5)	Safety-Related Valve Position Description	NOTE 3(a)	All	All	-	03/31/80	03/31/80
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	NOTE 3(a)	All	All	-	03/31/80	NA
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	NOTE 3(a)	NA	B&W	-	03/31/80	NA
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	NOTE 3(a)	NA	B&W	-	03/31/80	NA
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred Out of	NOTE 3(a)	All	All	-	03/31/80	NA

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
	Containment Inadvertently						
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	NOTE 3(a)	All	All	-	03/31/80	03/31/80
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading Up to, and in Early Phases of, the TMI-2 Accident	NOTE 3(a)	All	All	-	03/31/80	NA
II.K.1(12)	One-Hour Notification Requirement and Continuous Communications Channels	NOTE 3(a)	All	All	-		NA
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	NOTE 3(a)	All	All	-	01/01/81	01/01/81
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	NOTE 3(a)	GE	CE, W	-	03/31/80	NA
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	NOTE 3(a)	NA	CE, W	-	NA	
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	NOTE 3(a)	NA	CE, W	-	NA	
II.K.1(17)	Trip PZR Level Bistable So That PZR Low Pressure Will Initiate Safety Injection	NOTE 3(a)	NA	W	-		
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	NOTE 3(a)	NA	B&W	-	NA	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	NOTE 3(a)	NA	B&W	-	03/31/80	NA
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	NOTE 3(a)	NA	B&W	-	03/31/80	03/31/80
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	NOTE 3(a)	NA	B&W	-	03/31/80	03/31/80
II.K.1(22)	Describe Automatic and Manual Actions for Proper	NOTE 3(a)	All	NA	-	03/31/80	03/31/80

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
	Functioning of Auxiliary Heat Removal Systems When FW System Not Operable						
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	NOTE 3(a)	All	NA	-	03/31/80	03/31/80
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses between Reactor Trip and RCP Trip	NOTE 3(a)	NA	All	-	NA	
II.K.1(25)	Develop Operator Action Guidelines	NOTE 3(a)	NA	All	-	NA	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	NOTE 3(a)	NA	All	-	NA	
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	NOTE 3(a)	NA	All	-	NA	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	NOTE 3(a)	NA	All	-	01/01/81	01/01/82
II.K.2	Commission Orders on B&W Plants						
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	NOTE 3(a)	NA	B&W	-	NA	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	NOTE 3(a)	NA	B&W	-	NA	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	NOTE 3(a)	NA	B&W	-	NA	
II.K.2(4)	Small-Break LOCA Analysis, Procedures, and Operator Training	NOTE 3(a)	NA	B&W	-	NA	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	NOTE 3(a)	NA	B&W	-	NA	
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	NOTE 3(a)	NA	B&W	-	NA	
II.K.2(7)	Reevaluate Transient of September 24, 1977	NOTE 3(a)	NA	B&W	-	NA	
II.K.2(9)	Analysis and Upgrading of Integrated Control System	I	NA	B&W	F-27	01/01/81	01/01/81
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	I	NA	B&W	F-28	01/01/81	01/01/81
II.K.2(11)	Operator Training and Drilling	I	NA	B&W	F-29	01/01/81	01/01/81
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA with No AFW	I	NA	B&W	F-30	01/01/81	01/01/81
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	I	NA	B&W	F-31	01/01/81	01/01/81
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes after Primary System Voiding	I	NA	B&W	-	06/01/80	06/01/80

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power	I	NA	B&W	F-32	06/01/80	06/01/80
II.K.2(17)	Analysis of Potential Voiding in RCS during Anticipated Transients	I	NA	B&W	F-33	NA	
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	I	NA	B&W	F-34	01/01/81	NA
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	I	NA	B&W	F-35	01/01/81	NA
II.K.2(21)	LOFT L3-1 Predictions	NOTE 3(a)	NA	B&W	-	NA	
II.K.3	Final Recommendations of Bulletins and Orders Task Force						
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	I	NA	All	F-36	07/01/81	07/01/81
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	I	NA	All	F-37	01/01/81	01/01/81
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	I	All	All	F-38	04/01/80	04/01/80
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	I	NA	All	F-39, G-01	01/01/81	01/01/81
II.K.3(7)	Evaluation of PORV Opening Probability during Overpressure Transient	I	NA	B&W		01/01/81	01/01/81
II.K.3(9)	Proportional Integral Derivative Controller Modification	I	NA	W	F-40	07/01/80	07/01/80
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	I	NA	W	F-41		
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc., Until Further Review Complete	I	All	All	-		
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	I	NA	W	F-42	07/01/80	07/01/80
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	I	GE	NA	F-43	10/01/80	10/01/80
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	I	GE	NA	F-44	01/01/81	NA
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	I	GE	NA	F-45	01/01/81	01/01/81

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves—Feasibility Study and System Modification	I	GE	NA	F-46	01/01/81	01/01/81
II.K.3(17)	Report on Outage of ECC Systems—Licensee Report and Technical Specification Changes	I	GE	NA	F-47	01/01/81	01/01/81
II.K.3(18)	Modification of ADS Logic—Feasibility Study and Modification for Increased Diversity for Some Event Sequences	I	GE	NA	F-48	01/01/81	01/01/81
II.K.3(19)	Interlock on Recirculation Pump Loops	I	GE	NA	F-49	01/01/81	NA
II.K.3(20)	Loss of Service Water for Big Rock Point	I	GE	NA	-	01/01/81	NA
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level—Design and Modification	I	GE	NA	F-50	01/01/81	01/01/81
II.K.3(22)	Automatic Switchover of RCIC System Suction—Verify Procedures and Modify Design	I	GE	NA	F-51	01/01/81	01/01/81
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	I	GE	NA	F-52	01/01/82	01/01/82
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	I	GE	NA	F-53	01/01/82	01/01/82
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	I	GE	NA	F-54	10/01/80	10/01/80
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	I	GE	NA	F-55	01/01/82	01/01/82
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	I	GE	NA	F-56	04/01/81	NA
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	I	All	All	F-57	01/01/83	01/01/83
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	I	All	All	F-58	01/01/83	01/01/83
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	I	GE	NA	F-59	01/01/81	01/01/81
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	I	GE	NA	F-60	01/01/81	01/01/81
II.K.3(46)	Response to List of Concerns from ACRS Consultant	I	GE	NA	F-61	07/01/80	07/01/80
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	I	GE	NA	F-62	10/01/80	NA

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness—Short Term						
III.A.1.1	Upgrade Emergency Preparedness						
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	I	All	All	-	10/10/79	08/19/80
III.A.1.2	Upgrade Licensee Emergency Support Facilities						
III.A.1.2(1)	Technical Support Center	I	All	All	F-63	09/13/79	09/27/79
III.A.1.2(2)	On-Site Operational Support Center	I	All	All	F-64	09/13/79	09/27/79
III.A.1.2(3)	Near-Site Emergency Operations Facility	I	All	All	F-65	09/13/79	09/27/79
III.A.2	Improving Licensee Emergency Preparedness—Long Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E						
III.A.2.1(1)	Publish Proposed Amendments to the Rules	NOTE 3(a)	All	All	-		
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	I	All	All	F-67		
III.A.2.2	Development of Guidance and Criteria	I	All	All	F-68		
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.3	Communications						
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	NOTE 3(a)	All	All			
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	NOTE 3(a)	All	All	-		
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure						
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	I	All	All	-	07/02/79	09/27/79
III.D.3	Worker Radiation Protection Improvement						

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
III.D.3.3	Inplant Radiation Monitoring						
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	I	All	All	F-69	09/13/79	09/27/79
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	NOTE 3(a)	All	All	-	09/13/79	09/27/79
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	NOTE 3(a)	All	All	-	09/13/79	09/27/79
III.D.3.3(4)	Issue a Regulatory Guide	NOTE 3(a)	All	All	-	09/13/79	09/27/79
III.D.3.4	Control Room Habitability	I	All	All	F-70	05/07/80	06/26/80
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	NOTE 3(a)	All	All	-	NA	03/15/84
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	NOTE 3(a)	NA	All	D-10	01/81	01/81
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	W	-	04/17/85	04/17/85
A-4	CE Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	CE	-	04/17/85	04/17/85
A-5	B&W Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	B&W	-	04/17/850	4/17/85
A-6	Mark I Short-Term Program (former USI)	NOTE 3(a)	GE	NA	-	12/77	NA
A-7	Mark I Long-Term Program (former USI)	NOTE 3(a)	GE	NA	D-01	08/82	08/82
A-8	Mark II Containment Pool Dynamic Loads—Long Term Program (former USI)	NOTE 3(a)	GE	NA	-	08/81	08/81
A-9	ATWS (former USI)	NOTE 3(a)	All	All	-	06/26/84	06/26/84
A-10	BWR Feedwater Nozzle Cracking (former USI)	NOTE 3(a)	All	NA	B-25	11/80	11/80
A-11	Reactor Vessel Materials Toughness (former USI)	NOTE 3(a)	All	All	-	10/82	NA
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	NOTE 3(a)	NA	All	-	NA	TBD
A-13	Snubber Operability Assurance	NOTE 3(a)	All	All	B-17, B-22	1980	1980
A-16	Steam Effects on BWR Core Spray Distribution	NOTE 3(a)	GE	NA	D-12	NA	
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	NOTE 3(a)	All	All	B-60	08/81	08/81
A-25	Non-Safety Loads on Class 1E Power Sources	NOTE 3(a)	All	All	-	09/78	
A-26	Reactor Vessel Pressure Transient Protection	NOTE 3(a)	NA	All	B-04	09/78	09/78

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
	(former USI)						
A-28	Increase in Spent Fuel Pool Storage Capacity	NOTE 3(a)	All	All	-	04/17/78	NA
A-31	RHR Shutdown Requirements (former USI)	NOTE 3(a)	All	All	-	05/78	10/01/78
A-35	Adequacy of Offsite Power Systems	NOTE 3(a)	All	All	B-23	06/02/77	1980
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	NOTE 3(a)	All	All	C-10, C-15	07/80	07/80
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	NOTE 3(a)	GE	NA	-	02/29/80	09/30/80
A-40	Seismic Design Criteria (former USI)	NOTE 3(a)	All	All	-	TBD	09/89
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	NOTE 3(a)	All	NA	B-05	02/81	02/81
A-43	Containment Emergency Sump Performance (former USI)	NOTE 3(a)	NA	All	-	NA	11/85
A-44	Station Blackout (former USI)	NOTE 3(a)	All	All	-	TBD	06/88
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	NOTE 3(a)	All	All	-	02/87	NA
A-47	Safety Implications of Control Systems (former USI)	NOTE 3(a)	All	All	-	09/20/89	09/20/89
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	NOTE 3(a)	All	W	-	12/81	12/81
A-49	Pressurized Thermal Shock (former USI)	NOTE 3(a)	NA	All	A-21	TBD	07/85
B-10	Behavior of BWR Mark III Containments	NOTE 3(a)	GE	NA	-	NA	09/84
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	NOTE 3(a)	All	All	-	03/78	
B-56	Diesel Reliability	NOTE 3(a)	All	All	D-19	06/93	06/93
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	NOTE 3(a)	All	All	B-45	04/20/81	
B-64	Decommissioning of Reactors	NOTE 3(a)	All	All	-	06/27/88	NA
B-66	Control Room Infiltration Measurements	NOTE 3(a)	All	All	-	NA	07/81
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	NOTE 3(a)	All	All	-	05/27/80	05/27/80
C-10	Effective Operation of Containment Sprays in a LOCA	NOTE 3(a)	All	All	-	NA	
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	NOTE 3(a)	All	All	-	12/27/82	12/27/82

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
		NEW GENERIC ISSUES					
25	Automatic Air Header Dump on BWR Scram System	NOTE 3(a)	All	NA	-	01/09/81	01/09/81
40	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	NOTE 3(a)	All	NA	B-65	08/31/81	08/31/81
41	BWR Scram Discharge Volume Systems	NOTE 3(a)	All	NA	B-58	12/09/80	NA
43	Reliability of Air Systems	NOTE 3(a)	All	All	B-107	08/08/88	08/08/88
45	Inoperability of Instrumentation Due to Extreme Cold Weather	NOTE 3(a)	All	All	-	NA	09/01/83
51	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	NOTE 3(a)	All	All	L-913	07/18/89	07/18/89
67	Steam Generator Staff Actions						
67.3.3	Improved Accident Monitoring	NOTE 3(a)	All	All	A-17	12/17/82	12/17/82
70	PORV and Block Valve Reliability	NOTE 3(a)	NA	All	-	06/25/90	06/25/90
73	Detached Thermal Sleeves	NOTE 3(a)	NA	W	-	NA	
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	NOTE 3(a)	All	All	B-76, B-77, B-78, B-79, B-80, B-81, B-82, B-85, B-86, B-87, B-88, B-89, B-90, B-91, B-92, B-93	07/08/83	TBD
86	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	NOTE 3(a)	All	NA	B-84	TBD	TBD
87	Failure of HPCI Steam Line without Isolation	NOTE 3(a)	All	All	-	06/28/89	06/28/89
89	Stiff Pipe Clamps	MEDIUM	All	All	NA	NA	TBD
93	Steam Binding of Auxiliary Feedwater Pumps	NOTE 3(a)	NA	All	B-98	10/85	10/85
94	Additional Low Temperature Overpressure Protection for Light-Water Reactors	NOTE 3(a)	NA	CE, W	-	06/25/90	06/25/90
99	RCS/RHR Suction Line Valve Interlock on PWRs	NOTE 3(a)	NA	All	L-817	10/17/88	10/17/88

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
103	Design for Probable Maximum Precipitation	NOTE 3(a)	All	All	-	10/19/89	10/19/89
118	Tendon Anchorage Failure	NOTE 3(a)	All	All	NA	NA	07/90
124	Auxiliary Feedwater System Reliability	NOTE 3(a)	All	All	-	TBD	TBD
128	Electrical Power Reliability	NOTE 3(a)	All	All	-	04/29/91	04/29/91
130	Essential Service Water Pump Failures at Multiplant Sites	NOTE 3(a)	NA	All	-	09/19/91	09/19/91
155	Generic Concerns Arising from TMI-2 Cleanup						
155.1	More Realistic Source Term Assumptions	NOTE 3(a)	All	All	NA	NA	02/95
177	Vehicle Intrusion at TMI	NOTE 3(a)	All	All	-	08/01/94	08/01/94
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	ACTIVE	All	All	-	TBD	TBD
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion during A Severe Accident	ROI	All	All	-	TBD	TBD
191	Assessment of Debris Accumulation on PWR Sump Performance	ROI	NA	All	-	TBD	TBD
193	BWR ECCS Suction Concerns	ACTIVE	All	NA	-	TBD	TBD
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	ROI	All	All	-	TBD	TBD
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All	-	01/84	01/84